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QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH APRIL—JUNE 1978

MASTER

July 1978

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

DEPARTMENT OF ENERGY

IDAHO OPERATIONS OFFICE UNDER CONTRACT EY-76-C-07-1570

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Approved:

D. J. Olson, Manager
Semiscale Program

L. P. Leach, Manager
LOFT Experimental Program

H. J. Zeile, Manager
Thermal Fuels Behavior Program

Paul North
P. North, Manager
Code Development and Analysis Program

John A Dearien
J. A. Dearien, Manager
Code Verification and Applications Program

Richard E. Rice
R. E. Rice, Manager
3-D Experiment Project

L. J. Ybarrondo
L. J. Ybarrondo, Director
Water Reactor Research

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Approved by L. J. Ybarrondo
Technical Editor – E. L. Wills

July 1978

Idaho National Engineering Laboratory
Idaho Falls, ID 83401
operated by
EG&G Idaho, Inc.
for the
U.S. Department of Energy
Idaho Operations Office

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ABSTRACT

Water reactor research performed by EG&G Idaho, Inc., April through June 1978 is summarized for ongoing programs: Semiscale, LOFT, Thermal Fuels Behavior, Code Development and Analysis, Code Verification and Applications, and the 3-D Project. The Semiscale Program reports the performance of the first two tests in the new Mod-3 system and an evaluation of two LOFT counterpart tests, one (Test S-06-3) an NRC standard problem. The LOFT Experimental Program reports significant results from the final test in the nonnuclear test series and summarizes the objectives of the first LOFT nuclear test series. The Thermal Fuels Behavior Program reports on the various test series in the PBF reactor (including results of a power-cooling-mismatch test), and presents discussions of failed fuel rod behavior and vapor explosion criteria. The Code Development and Analysis Program reports progress in developing an improved version of the RELAP4 code and a successful linking of two codes, RELAP4 (a blowdown reflood code) and FRAP-T (a fuel analysis code). The Code Verification and Applications Program reports progress in verifying the RELAP4/MOD6 and FRAP-T4 codes and describes the NRC/RSR Data Bank Program, which is intended to collect, store, and provide access to experiment data on reactor safety. The multinational 3-D Experiment Project reports design, analysis, and instrumentation development in support of German and Japanese reflood experiments.

PREFACE

EG&G Idaho, Inc., performs technical activities in the water reactor safety programs at the Idaho National Engineering Laboratory under the sponsorship of the U. S. Nuclear Regulatory Commission's Division of Reactor Safety Research. The current water reactor research activities of EG&G Idaho, Inc., are accomplished in five programs: the Semiscale Program, the Loss-of-Fluid Test (LOFT) Experimental Program, the Thermal Fuels Behavior Program, the Code Development and Analysis Program and the Code Verification and Applications Program (which were formerly the Reactor Behavior Program), and the 3-D Experiment Project.

The Semiscale Program consists of a continuing series of small-scale nonnuclear thermal-hydraulic experiments having as their primary purpose the generation of experiment data that can be applied to the development and verification of analytical models describing loss-of-coolant accident (LOCA) phenomena in water-cooled nuclear power plants. Emphasis is placed on acquiring system effects data from integral tests that characterize the most significant thermal-hydraulic phenomena likely to occur in the primary coolant system of a nuclear plant during the depressurization (blowdown) and emergency cooling phase of a LOCA. The recently completed program of experiments employing the Semiscale Mod-1 test system, that used one intact loop with active components and a broken loop with passive components, has included core reflood and emergency core cooling tests using an electrically heated 40-rod core. The Semiscale test facility has been converted to a new test system (Mod-3) that contains two active loops and a full-length core and is scaled more directly to a pressurized water reactor (PWR).

The LOFT Experimental Program is a nuclear test program for providing test data to support (a) assessment and improvement of the analytical methods utilized for predicting the behavior of a PWR under LOCA conditions, (b) evaluation of the performance of PWR engineered safety features, particularly the emergency core cooling system, and (c) assessment of the quantitative margins of safety inherent in the performance of these safety features. The test program utilizes the LOFT Facility, an extensively instrumented 55-MW pressurized water reactor facility designed to conduct loss-of-coolant experiments (LOCEs). The test program includes a series of nonnuclear (without nuclear heat) LOCEs followed by a series of low-power nuclear LOCEs and then a series of high-power nuclear LOCEs.

The Thermal Fuels Behavior Program is an integrated experimental and analytical program designed to provide information on the behavior of reactor fuels under normal, off-normal, and accident conditions. The experiment portion of the program is concentrated on testing of single fuel rods and fuel rod clusters under power-cooling-mismatch, loss-of-coolant, and reactivity initiated accident conditions. These tests provide in-pile experiment data for the evaluation and verification of analytical models that are used to predict fuel behavior under reactor conditions spanning normal operation through severe hypothetical accidents. Data from this program provide a basis for improvement of the fuel models.

The earlier Reactor Behavior Program has been realigned as two programs dealing with code development and with code verification. The Code Development and Analysis Program has the primary responsibility for the development of codes and analysis methods; it provides the analytical research aimed at predicting the response of nuclear power reactors under normal, off-normal, and accident conditions. The Code Verification and Applications Program performs the task of verifying the accuracy and range of applicability of computer codes developed for the analysis of reactor behavior. The verification process involves the analyses of many different experiments and the comparison of calculated results with experimental data. Statistical evaluation of both the analytical and experimental results are part of the verification process.

The 3-D Experiment Project is a multinational cooperative water reactor research project which is designed to study the behavior of entrained liquid in a full-scale upper plenum and cross flow in the core during the reflood phase of a PWR LOCA.

A more detailed description of the first four programs is presented in the quarterly report for January through March 1975, ANCR-1254. Later quarterly reports are ANCR-1262 (for April–June 1975), ANCR-1296 (for July–September 1975), ANCR-NUREG-1301 (for October–December 1975), ANCR-NUREG-1315 (for January–March 1976), TREE-NUREG-1004 (for April–June 1976), TREE-NUREG-1017 (for July–September 1976), TREE-NUREG-1070 (for October–December 1976), TREE-NUREG-1128 (for January–March 1977), TREE-NUREG-1147 (for April–June 1977), TREE-NUREG-1188 (for July–September 1977), TREE-NUREG-1205 (for October–December 1977), and TREE-NUREG-1218 (for January–March 1978). Copies of the quarterly reports are available from the Technical Information Center, Department of Energy, Oak Ridge, Tennessee and the National Technical Information Service, Springfield, Virginia.

SUMMARY

Water reactor research activities performed by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory for April through June 1978 are reported for the Semiscale Program, the LOFT Program, the Thermal Fuels Behavior Program, the Code Development and Analysis Program and the Code Verification and Applications Program (which were the Reactor Behavior Program), and the 3-D Experiment Project.

For the Semiscale Program, major effort at the Semiscale test facility included the performance of the first two tests using the Semiscale Mod-3 system. Test S-07-4, the initial test of the Mod-3 baseline test series (Test Series 7), and Test S-07-1 were conducted in June 1978. Preliminary results indicate that test objectives were met and that the data from the two tests will improve understanding of Mod-3 system response and core reflood behavior. Results of two Semiscale Mod-1 tests are evaluated; the tests were conducted to determine the effect of special operating conditions on system response. Tests S-06-3 and S-06-6 were conducted as 200% cold leg break experiments to simulate the 75% of full power test planned for the first LOFT nuclear test series (Series L2). Test S-06-3 was conducted as U.S. Nuclear Regulatory Commission Standard Problem 8. Preliminary analysis of Tests S-06-3 and S-06-6 indicates that Mod-1 system response during decompression was sensitive to the special operating conditions and that the system during refill and reflood was sensitive to decompression time and accumulator nitrogen injection initiation.

The LOFT Program completed the final loss-of-coolant experiment (LOCE) in the LOFT nonnuclear test series. For this experiment (LOCE L1-5), the first LOFT nuclear core (Core 1) and associated instrumentation were installed; however, the nuclear reactor was held in a shutdown condition. LOCE L1-5 was an isothermal blowdown, simulating a double-ended offset shear on the inlet side of one of the primary coolant loops of a four-loop pressurized water reactor. Emergency core coolant (ECC) was injected into the intact loop cold leg during the

blowdown. The data from LOCE L1-5 show the reactor vessel downcomer to void of fluid relatively early in the blowdown and the reactor vessel lower plenum to only partially void of fluid throughout the blowdown. Asymmetric fluid flow occurred in the downcomer during ECC injection and in the reactor core during refill of the lower plenum. Fuel rod cladding temperatures increased due to low core steam flow during refill of the lower plenum.

LOFT Test Series L2, scheduled to be started in late 1978, is the first series of LOCEs to be conducted in LOFT with the nuclear core generating heat. Test Series L2 is a series of five LOCEs that will be conducted to examine the response of a nuclear reactor to controlled pipe ruptures in the primary coolant system cold leg piping. Each LOCE will simulate a 200% double-ended offset shear in the primary coolant cold leg piping. The third LOCE in the L2 series will be run at full power (52.5 kW/m), which is consistent with the primary objective to progress to the full-power LOCE as early in the series as possible without jeopardizing the integrity of the LOFT test series.

Thermal Fuels Behavior Program accomplishments included performance of a power-cooling-mismatch (PCM) test (Test PCM-5) and five driver core reactivity-initiated-accident (RIA) lead rod tests. Results were reported from tests previously performed in the PCM, Loss-of-Coolant Accident (LOCA), and Gap Conductance (GC) Test Series. Work continued on the Halden Fuel Behavior Research Program and the Power Reactor Post-irradiation Examination Program. A major accomplishment in the RIA Test Series was the completion of the RIA Scoping Test experiment predictions. Analysis of gap conductance test data was performed for Tests GC 2-1, GC 2-2, and GC 2-3. A method was developed for modifying the Ross and Stoute correlation for gap conductance to account for pellet cracking and fuel fragment relocation. Results of Test PCM-1 show that cladding surface temperature measurement data are in good agreement with the FRAP-T3 calculations. Development and evaluation efforts were performed in the areas of PBF program development, coordination with foreign experimental programs, Nuclear Regulatory Commission technical assistance, analysis of test results, comparison of vapor explosion criteria for various analytical models, Halden fuel behavior research, and postirradiation examination of commercial power reactor fuel.

The Code Development and Analysis Program accomplishments were made in reference code development, fuel analysis research, and advanced code development. Primary model development efforts were directed toward development of RELAP4/MOD7, the integral blowdown/reflood code, to provide a fast-running, user convenient code package that includes integral LOCA analysis capability and has improved modeling capabilities over earlier code versions. Among the RELAP4/MOD7 improvements are an automatic self-initialization feature that incorporates an energy balance model (to ensure that the total system net heat transfer rate is zero) and a pressure balance model (to ensure that all control volume thermodynamic pressures are consistent with input relative to flow rate, geometry, and resistance to flow). Another significant improvement is the successful linking of the FRAP-T fuel analysis program and the RELAP4 code.

The Code Verification and Applications Program progressed in assessing and verifying the RELAP4/MOD6 thermal-hydraulic code and the FRAP-T4 fuel analysis program by comparing calculations using the codes with experimental data from numerous test facilities, including Semiscale, THTF, FLECHT, and the West German PKL facility. In the area of technical surveillance of NRC/industry cooperative programs, comparisons have been made between RELAP4/MOD6 code calculations and experiment data from TLTA-3 Tests 6004 and 6005 of the boiling water reactor-blowdown/emergency core cooling program. The NRC/RSR Data Bank Program has been instituted to provide the means for collecting, processing, and providing access to reactor safety experiment data. The program uses a data bank processing system that is an expanding collection of computer programs that have been developed to provide the capability to accept data from established data sources, to output the data to tape in a standard format, and to allow on-line retrieval and comparison of the data.

The 3-D Experiment Project has completed design and begun fabrication of instrumented spool pieces and liquid level detectors for the JAERI reflood system experiment. Project planning has been completed for design of flow measuring devices for Japanese and German reflood experiments. Air-water upper plenum testing is being continued to provide data on flow mechanisms during the reflood phase of a LOCA. Results of air-water tests to date indicate that observed flooding behavior is geometry dependent and that, while linear behavior observed in classical flooding experiments does occur, the results differ markedly from those calculated by empirical correlations based on classical flooding experiments.

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I. SEMISCALE PROGRAM

D. J. Olson, Manager

The first two tests with the Semiscale Mod-3 system[a] have been completed, and preliminary results indicate that all test objectives were met and that the test data will significantly improve understanding of core reflood behavior.

An evaluation is presented of results of two Semiscale Mod-1 tests that were conducted to determine the effect of special operating conditions on system response. Tests S-06-3 and S-06-6 were conducted as 200% cold leg break experiments to simulate the 75% of full power test planned for the first LOFT nuclear test series (Series L2). Test S-06-3 was conducted as Nuclear Regulatory Standard Problem 8; the test data were released April 28, 1978.

1. SEMISCALE MOD-3 TESTING

The first two experiments in the Semiscale Mod-3 system were conducted during June 1978. Test S-07-4 was the initial test of the Semiscale Mod-3 baseline test series (Test Series 7) and was designed as a

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[a] The Semiscale Mod-3 system was discussed in earlier quarterly reports[1,2] and is described in detail in Reference 3.

gravity feed reflood test with initial conditions similar to the initial conditions of experiments conducted in both the Semiscale Mod-1 and FLECHT-SET facilities. The principal objectives of Test S-07-4, in addition to obtaining basic information about the Mod-3 system reflooding characteristics, were to investigate the reproducibility of results obtained from different systems and to help establish the influence of core length and broken loop components on the system reflood response.

The objectives of Test S-07-1 were to provide information on the effects of the Mod-3 system changes (relative to Mod-1) on basic system thermal-hydraulic behavior during blowdown. Test S-07-1 was performed with initial conditions and a core power decay that approximated those of a previous experiment with the Semiscale Mod-1 system (Test S-02-9[4]). Performance of identical tests in both systems is intended to allow isolation of the system thermal-hydraulic behavior caused by differences between the two test facilities. Two further objectives of Test S-07-1 were (a) to investigate the basic downcomer emergency core coolant (ECC) penetration characterisitcs of the Mod-3 system by injecting ECC into the intact loop cold leg and (b) to determine core thermal conditions prior to core reflood; this information is needed for future core reflood tests.

A preliminary evaluation of the data from Tests S-07-4 and S-07-1 indicates that the important test parameters were within the specified tolerances and that the objectives were met. The in-core instrumentation unique to the Mod-3 system appears to have supplied results which will significantly improve the understanding of core reflood behavior.

2. EFFECT OF SPECIAL HARDWARE OPERATING ASSUMPTIONS DURING SEMISCALE MOD-1 TESTING

C. E. Cartmill

This section presents an evaluation of the results of two Semiscale Mod-1 tests which were conducted to determine the effect on system response of a set of special hardware operating conditions (assumptions). This set of operating conditions included assumptions on the operation of the pump, the steam generator, the pumped emergency core cooling injection system, and the containment simulation. Variations in operation of the hardware were selected because of the potentially large influence of these components on system response during the blowdown through reflood phases of a possible loss-of-coolant accident (LOCA) in a PWR plant.

The two tests in the Semiscale Mod-1 system were 200% cold leg break experiments conducted at an axial peak power density of 41.0 kW/m on the high power rods (simulating the 75% power test currently scheduled by the LOFT program). The core radial power profile was peaked in each test to simulate the radial peaking in the LOFT nuclear core, and ECC was injected into the intact loop cold leg only. Table I lists the corresponding special hardware assumptions for Test S-06-6 and for Test S-06-3. The remaining test operation specifications were the same for each test. Both tests were conducted using the Semiscale-LOFT counterpart break nozzles that are designed geometrically similar to the nozzles used in the LOFT system.

The influence of each of the assumptions on system response is discussed with respect to the blowdown through reflood phases of the tests. In discussing the test results as a function of time, the influence of each hardware assumption is treated as it occurs during the test, providing both a "hardware" effect and "time sequence" effect on system response.

TABLE I

SPECIFIED DIFFERENCES BETWEEN TESTS S-06-3 AND S-06-6 RESULTING FROM HARDWARE CONDITIONS

System or Component	Test S-06-3	Test S-06-6
Intact loop pump	Power maintained	Power lost at rupture (coastdown to 30% by 20 seconds)
Broken loop pump	Simulated operating pump resistance K = 8.97	Simulated locked rotor pump resistance K = 12.72
Steam generator steam valve	Remained open 8 seconds then ramped closed	Ramped closed in 15 second:
Steam generator feedwater valve	Remained open 8 seconds then ramped closed by 22 seconds	Ramped closed in 15 second:
High pressure injection system (HPIS)	Simulated HPIS with 2 pumps. Started at 12 411 kPa 2 pumps, 0.049 l/s	Simulated HPIS with 1 pump. Started after 25 seconds delay, 0.031 l/s
Low pressure injection system (LPIS)	Simulated LPIS with 2 pumps. Started at 1551.3 kPa, 0.297 l/s	Simulated LPIS with 1 pump started after 35 second delay, 0.208 l/s.
Containment pressure	248.2 kPa	196.5 kPa

2.1 Blowdown Phase

2.1.1 Overall System Response. Under identical system configuration and conditions, the Mod-1 system decompression and break flow rates following rupture are influenced most strongly by the cold and hot leg fluid temperatures. Since these temperatures were approximately the same for the two tests, the decompression was very similar up to the time containment pressure was reached, as shown in Figure 1. This similarity indicates no significant overall effect on system depressurization due to the special hardware assumptions. Likewise, the similarity of the break flow rates shown in Figure 2 indicates the hardware assumptions had no significant influence on break flow rates. The hot leg break flow rate was almost identical for the two tests indicating the hardware assumption of an increased broken loop simulated pump resistance has only a small effect on the hot leg break flow. The cold leg break demand supplied by the intact loop differed in each test as indicated by the mass flow out the cold leg break. The difference in total flow out the intact loop cold leg is attributed to the difference

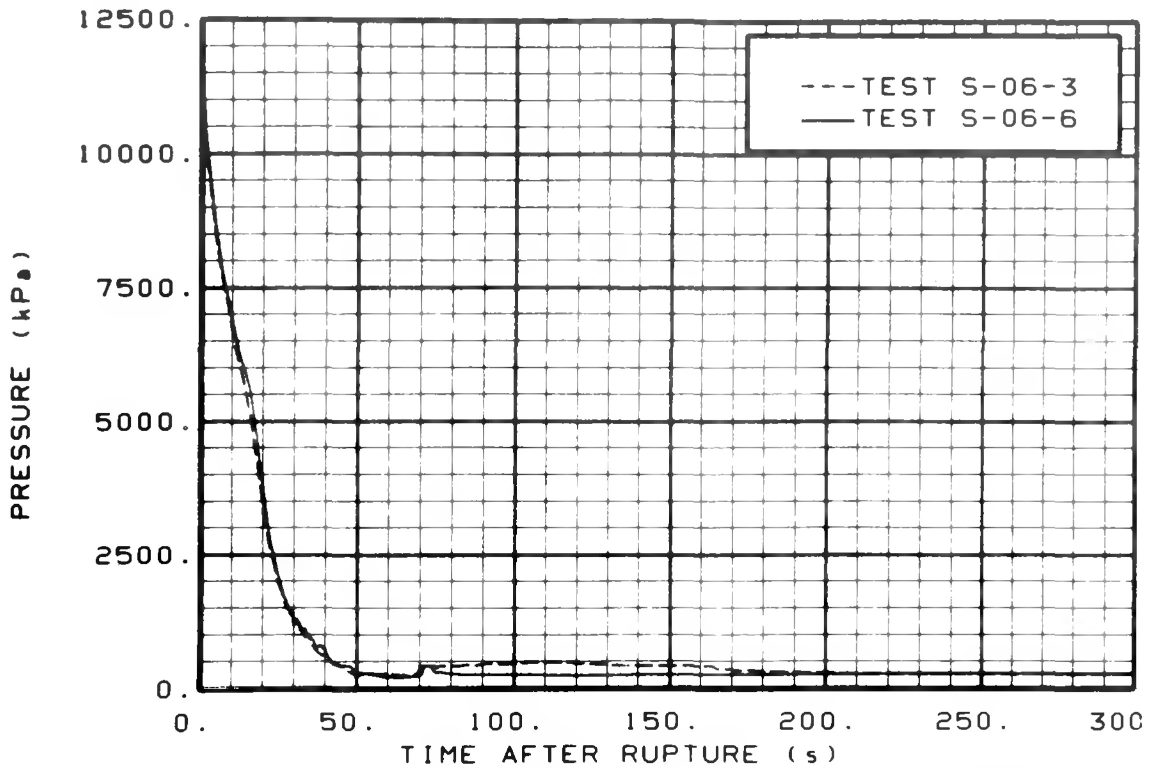


Fig. 1 Pressure in upper plenum, Tests S-06-3 and S-06-6 (PV-UP-10).

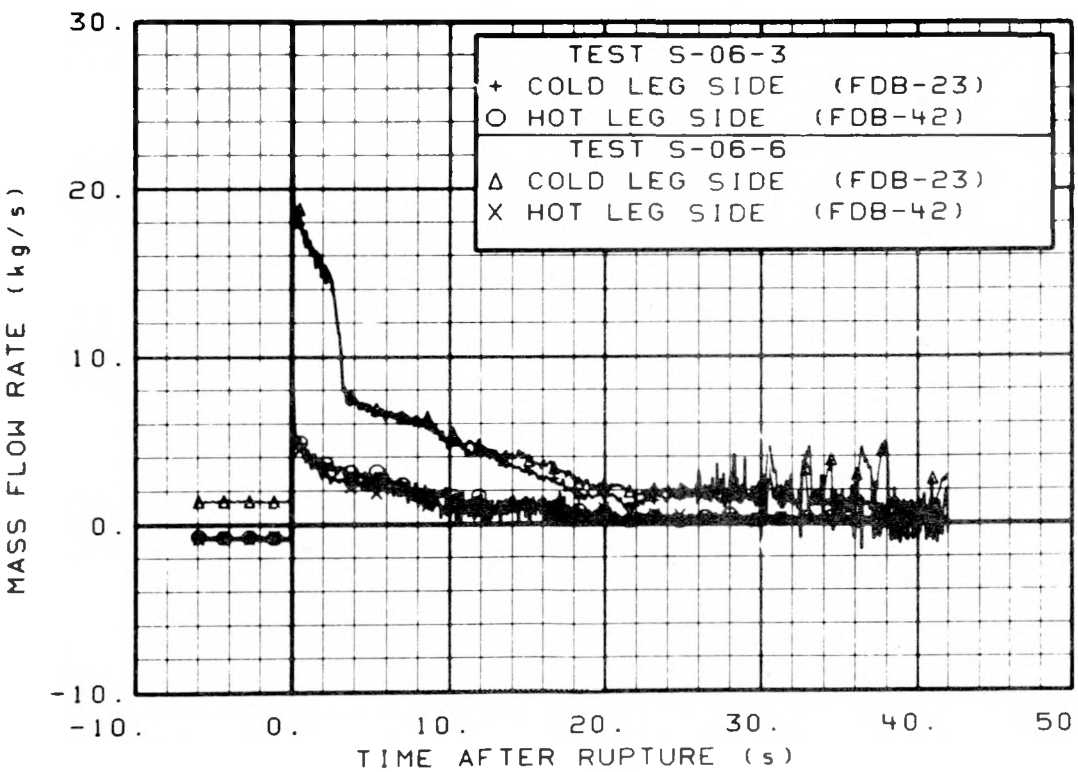


Fig. 2 Hot and cold leg break flow rates, Tests S-06-3 and S-06-6.

in active pump operation (Figure 3) which in Test S-06-3 remained on at 100% power and in Test S-06-6 coasted down as specified by the hardware assumptions. The effect of pump operation on integrated flow was similarily illustrated for other Semiscale integral blowdown tests (Tests S-05-2, S-05-2A, and S-05-2B). These other tests differed only in pump speed and indicated similar trends in integrated flow as were observed in Tests S-06-3 and S-06-6. The net result, therefore, is that in the special hardware assumption test (Test S-06-6) the core must supply a larger percentage of the cold leg break flow demand than in Test S-06-3, thus resulting in a sustained high core flow reversal subsequent to 2.5 seconds after rupture, as shown in Figure 4. This core flow behavior had a significant effect on the temperature of the rod cladding surfaces during blowdown.

2.1.2 Core Rod Behavior. Information for Tests S-06-3 and S-06-6 regarding average core rod behavior is given in Table II. The majority of the rod measurements at and above the 71-cm core elevation, with the exception of high powered rods, experienced a rewet for Test S-06-6 between 10 and 20 seconds following rupture. This rewet phenomenon is

TABLE II

AVERAGE BEHAVIOR OF CORE RODS IN TESTS S-06-3 and S-06-6

DNB Time After Rupture(s)/% All Rods		Quench Time(s)		Rewet		Maximum Cladding Temperature (K)		Quench Temperature (K)	
S-06-3	S-06-6	S-06-3	S-06-6	S-06-3	S-06-6	S-06-3	S-06-6	S-06-3	S-06-6
0.69/7% 3.87/93%	0.6/4% 4.09/96%	131	230	8%[a]	29%[b]	871	842	675	687

[a] 8% of all rods rewetted (only 15% of rods above 71-cm elevation).

[b] 29% of all rods rewetted (62% of rods above 71-cm elevation).

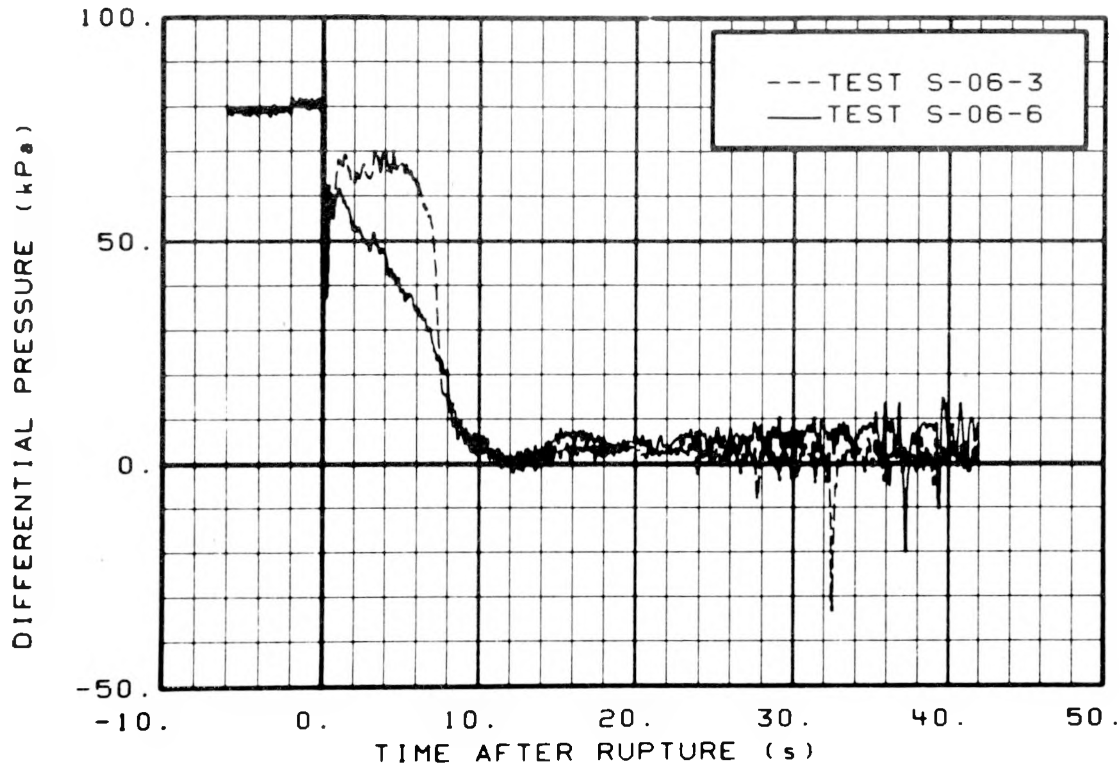


Fig. 3 Pump differential pressure, Tests S-06-3 and S-06-6 (DPU-12-10).

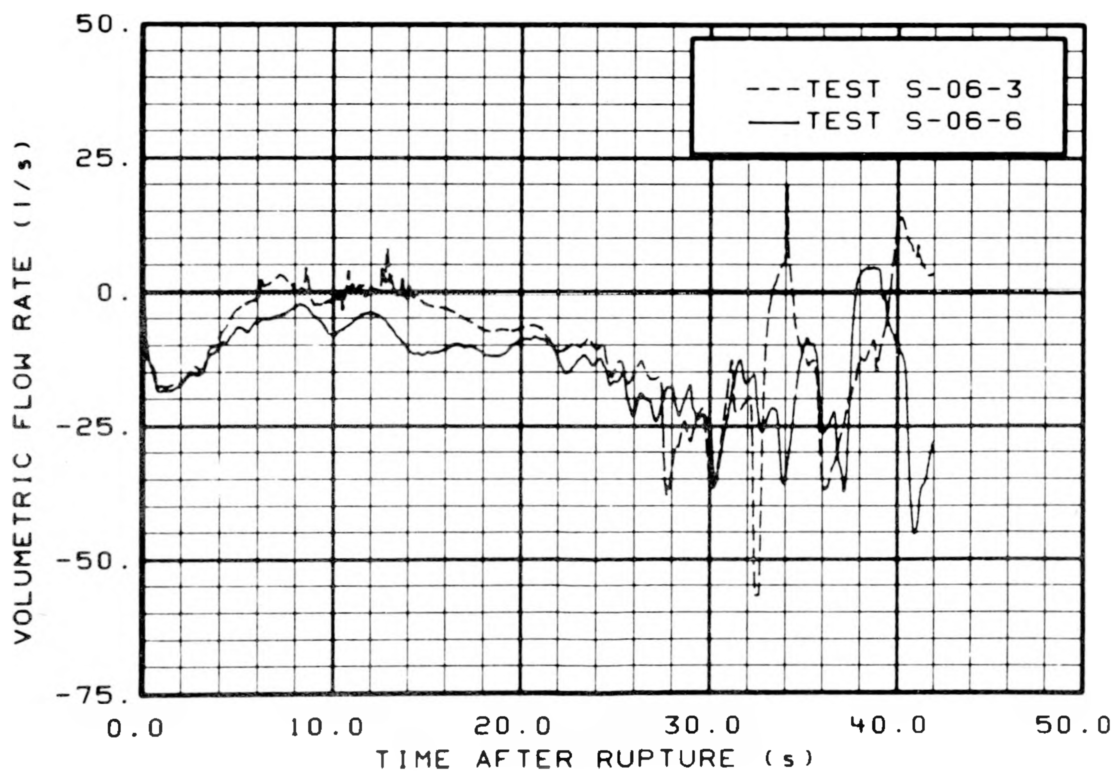


Fig. 4 Core inlet volumetric flow rate, Tests S-06-3 and S-06-6 (FTV-CORE-IN).

illustrated at three different axial elevations in Figure 5 and is thought to be the result of the high reverse core flow (Figure 4) which resulted from the use of the special hardware assumption of pump performance. Although those locations that had rewetted had lower temperatures during blowdown than those few locations that did not rewet, the rewet was only temporary because of the apparent core dryout that followed at about 20 seconds. Figure 6 compares the temperature on a rod that had rewetted with the temperature on a rod that did not rewet. As shown, the core dryout at about 20 seconds resulted in poor heat transfer and a rapid heating of the rods that had rewetted and caused the maximum temperature reached and the final quench time recorded in each case to be about the same. Nevertheless, a general reduction in maximum cladding temperatures occurred in Test S-06-6 as compared to Test S-06-3 due solely to the high core flow rate caused by the special hardware assumptions.

2.2 Refill and Reflood Phases

The general behavior of the Mod-1 system during refill and reflood is considered in view of the special hardware assumptions specified. Discussed first are the events leading to the initiation of lower plenum refill; a general discussion of the core reflood behavior is then presented. Figure 7 shows that the system in Test S-06-6 depressurized to the suppression tank pressure by about 58 seconds and lower plenum refill began about 10 seconds later, coinciding with the time that nitrogen injection from the intact loop accumulator started[a]. These results are similar to those observed in Test S-06-3. However, as shown in Figure 8, core reflood began at about 71 seconds in Test S-06-3, but did not begin until 82 seconds in Test S-06-6. This difference in reflood time is attributed primarily to the degree of lower plenum liquid depletion in each test at the onset of nitrogen injection from the accumulator and also to differences in the liquid inventory in the

[a] The 10-second delay from the time the system depressurizes to the time lower plenum refill begins is a reflection of the delay time associated with the Mod-1 hot downcomer walls.

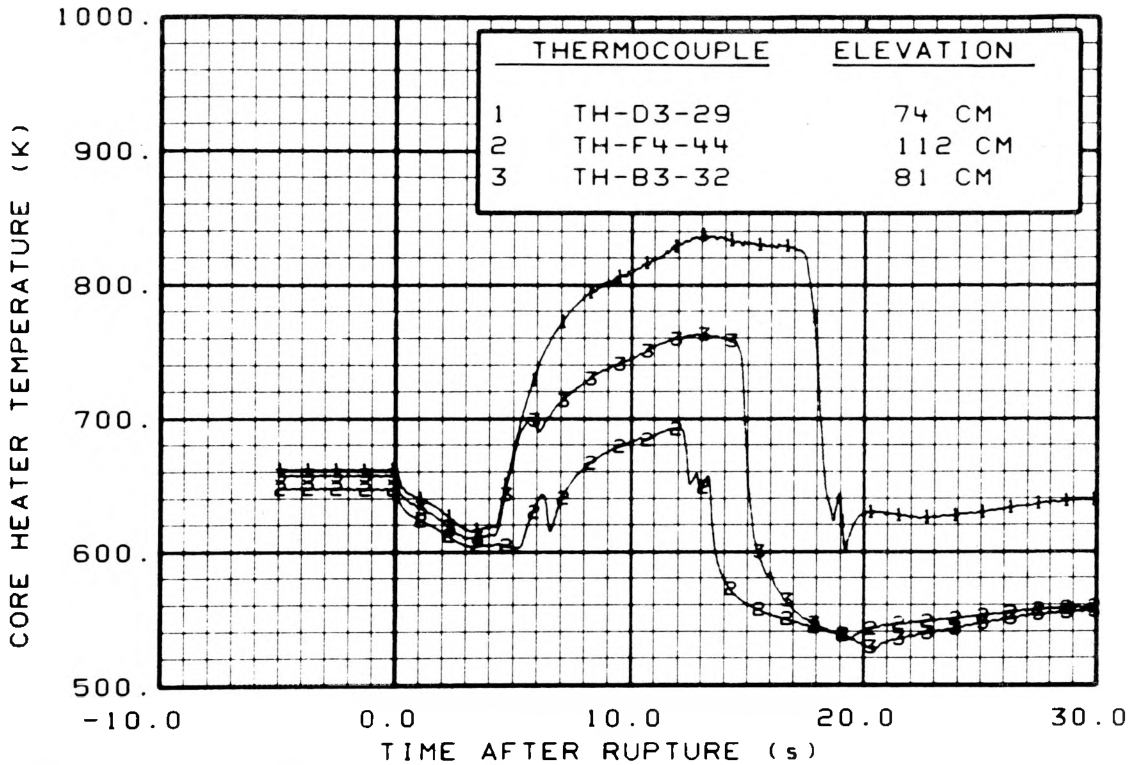


Fig. 5 Peak cladding temperatures at various elevations, Test S-06-6 (Thermocouples TH-D3-29 at 74 cm, TH-F4-44 at 112 cm, and TH-B3-32 at 81 cm above bottom of core).

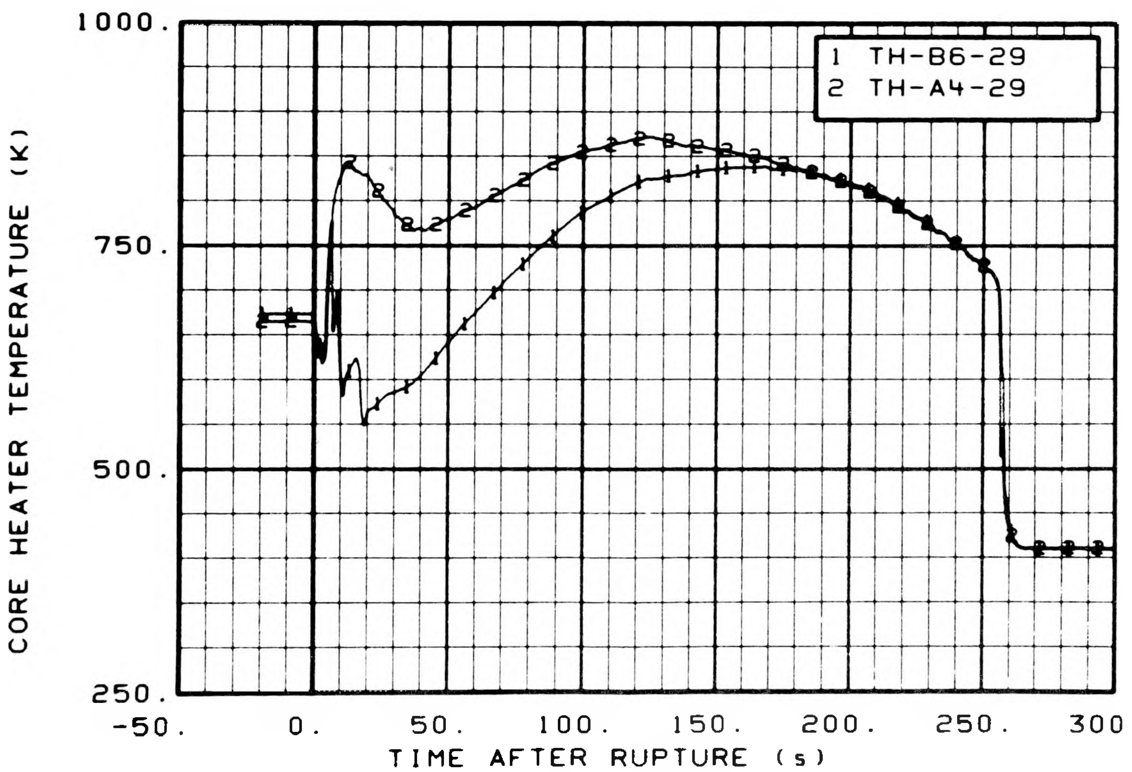


Fig. 6 Effect of rewet on peak cladding temperature of two rods, Test S-06-6 (Thermocouples TH-B6-29 and TH-A4-29 at 74 cm above bottom of core).

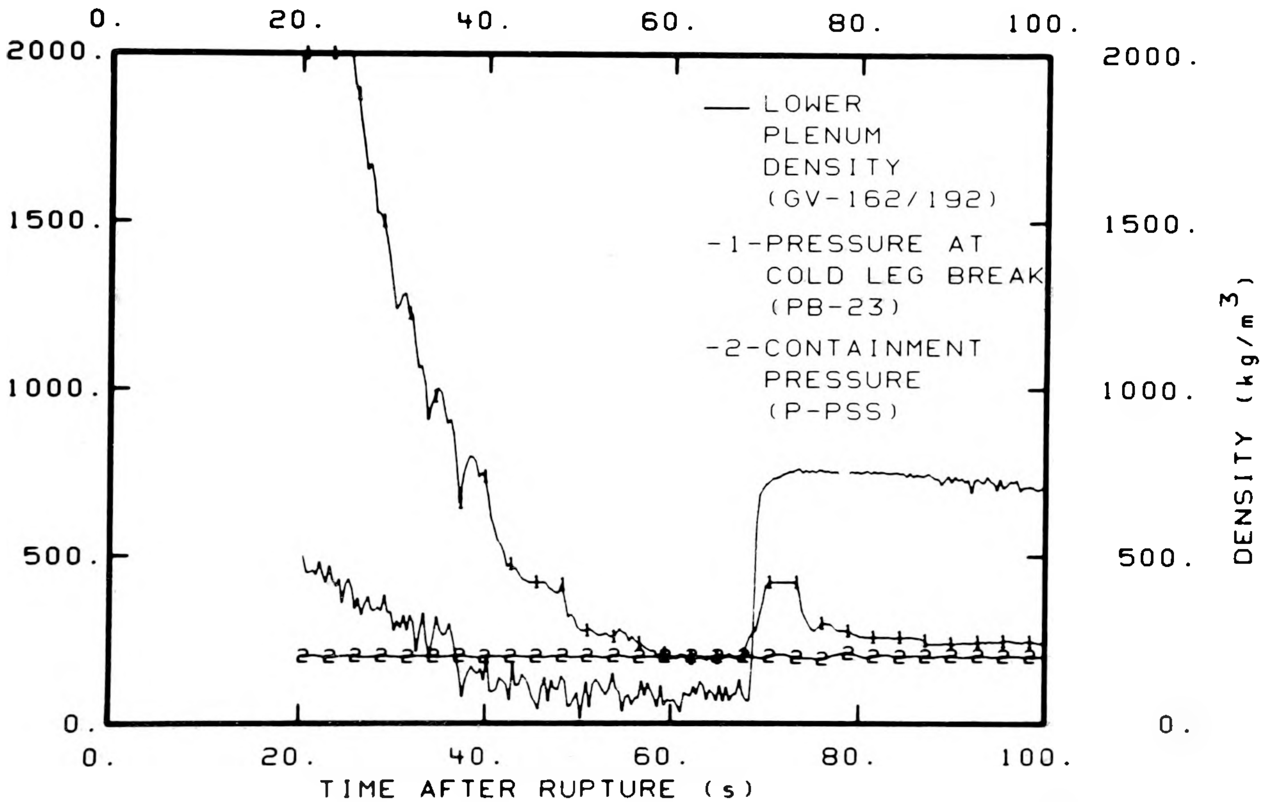


Fig. 7 System pressure, containment pressure, and lower plenum density for Test S-06-6.

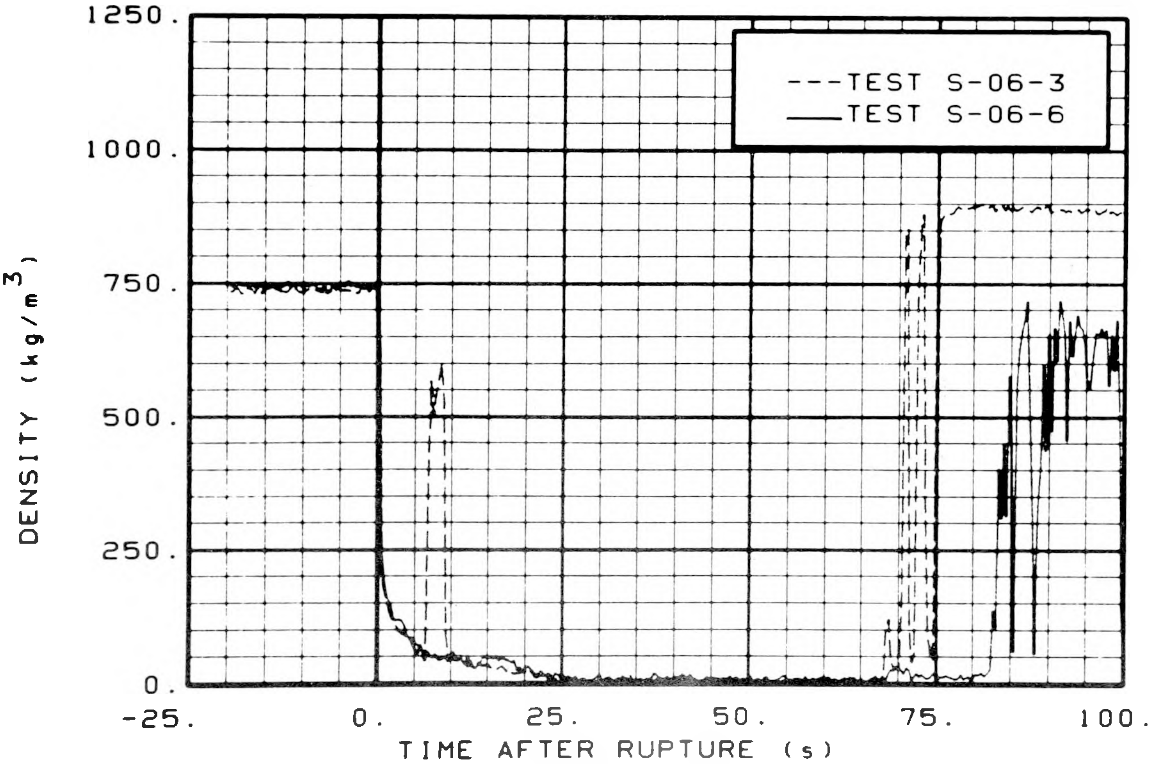


Fig. 8 Core inlet fluid density, Tests S-06-3 and S-06-6 (GV-COR-15HZ).

upper annulus and downcomer which falls to the lower plenum immediately following nitrogen injection. As shown in Figure 7, the system during Test S-06-6 did not reach containment pressure until 58 seconds with refill beginning at 68 seconds. A slight delay in the time to reach containment pressure in Test S-06-6 relative to Test S-06-3 was influenced by the slightly lower magnitude in containment pressure specified as part of the special hardware assumptions applied (Table I). Nevertheless, the significant fact is that in Test S-06-6 lower plenum refill was initiated by nitrogen injection from the intact loop accumulator and began prior to the end of the hot wall delay time or before end of bypass. As shown in Figure 9, which compares the lower plenum liquid levels for each test, the lower plenum in the case of Test S-06-3 had begun to refill prior to nitrogen injection from the accumulator and the liquid level had reached 26-cm above the bottom of the core at the time the accumulator began injecting nitrogen. The subsequent result during Test S-06-3 was a significant and prolonged surge in the overall vessel liquid level when the nitrogen pushed the liquid. For

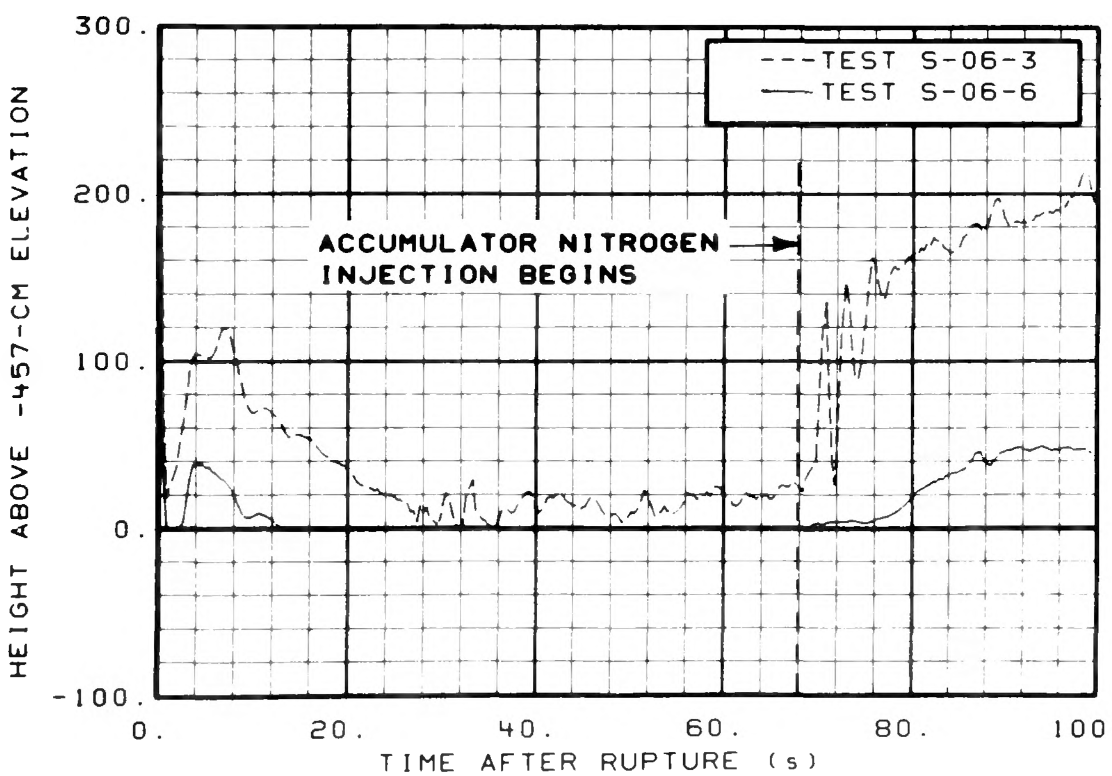


Fig. 9 Core collapsed liquid level (height above -457-cm elevation, 15 cm below bottom of core), Tests S-06-3 and S-06-6.

Test S-06-6, however, end of bypass had not been reached prior to nitrogen injection, and the relatively low inventory of water in the annulus and downcomer (coupled with the relatively lower plenum liquid level) resulted in a liquid level surge reaching to only the upper elevations of the lower plenum and delaying reflood, relative to Test S-06-3, by 11 seconds. Furthermore, after the nitrogen induced level surge took place in Test S-06-6, the remainder of refill was accomplished by the low pressure ECC injection system (LPIS) and high pressure ECC injection system (HPIS) flows which were less than in Test S-06-3 (prior to 88 seconds).

The rod quench times during reflood for Test S-06-3 and Test S-06-6 are indicated in Table II and the measured peak cladding temperature response throughout the transient is given in Table II. As indicated by the quench times, the quench front progressed at a much faster rate in Test S-06-3 than in Test S-06-6. Although the special hardware assumptions resulted in specification of low HPIS and LPIS flow rates in Test S-06-6, a malfunction of the HPIS system resulted in an average HPIS flow rate of about 0.145 l/s versus the 0.21 l/s specified. Consequently, the ECC injection rates during reflood were essentially the same in each test. The higher reflood rate in Test S-06-3 was apparently influenced primarily by the initial surge of liquid at the initiation of refill, which indicates that the Mod-1 system was extremely sensitive to the coupled phenomena following system decompression and accumulator nitrogen injection.

A preliminary analysis of the results from Tests S-06-3 and S-06-6 has led to the following conclusions with regard to the special hardware assumptions.

(1) The Mod-1 system response during blowdown was shown to be sensitive to the hardware operating assumptions applied. In particular, the assumptions governing active pump operation and simulated pump hydraulic resistance had a substantial effect on the core hydraulics and core heat transfer. In

Test S-06-6 lower peak cladding temperatures resulted relative to Test S-06-3.

(2) The Mod-1 system response during refill and reflood was shown to be extremely sensitive to the system decompression time and the time at which nitrogen injection from the accumulator started. These times were different for Test S-06-3 compared to Test S-06-6 because of problems encountered with the ECC injection system. Therefore, no definite conclusions could be reached regarding the effect of the special hardware assumptions on the Mod-1 system response during this period of the test.

II. LOFT EXPERIMENTAL PROGRAM

L. P. Leach, Manager

The LOFT Program completed the final loss-of-coolant experiment (LOCE) in the LOFT nonnuclear test series. For this experiment (LOCE L1-5), the first LOFT nuclear core (Core 1) and associated instrumentation were installed; however, the nuclear reactor was held in a shutdown condition. LOCE L1-5 was an isothermal blowdown with emergency core coolant (ECC) injection into the intact loop cold leg. LOCE L1-5 provided data for a final evaluation of the LOFT facility prior to the first LOFT nuclear test series (Test Series L2).

Requalification of the LOFT test system following LOCE L1-5 is currently in progress. After the system is requalified, a series of nuclear power range tests will be performed in preparation for Test Series L2.

A summary of the more significant results from LOCE L1-5 and a summary of the Test Series L2 objectives are presented in the following sections.

1. LOFT LOSS-OF-COOLANT EXPERIMENT L1-5 RESULTS

P. G. Prassinos and A. C. Peterson

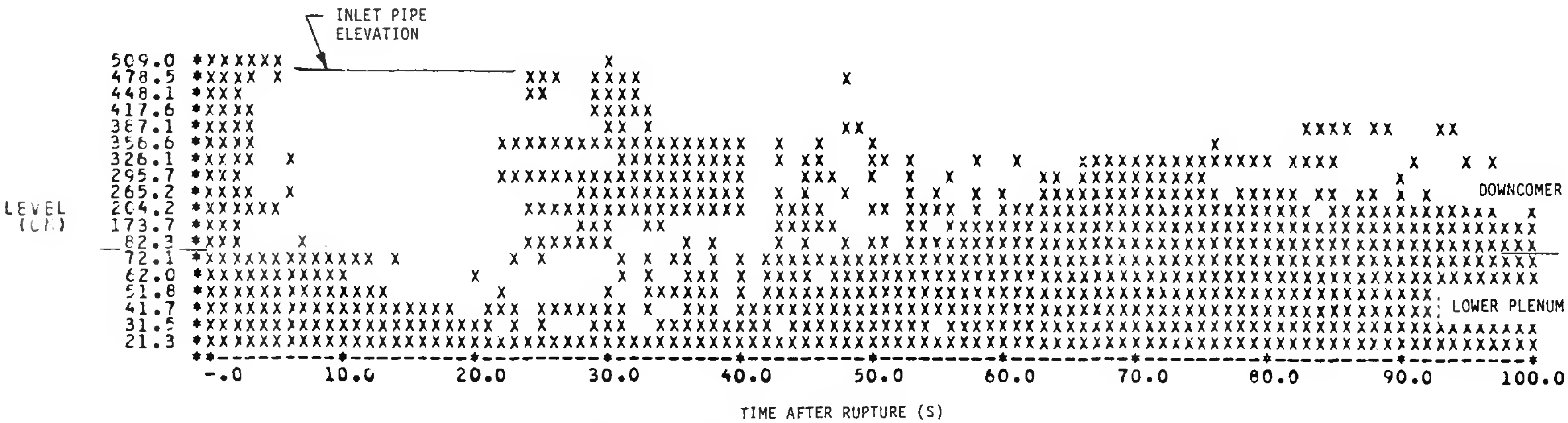
LOFT LOCE L1-5 simulated a double-ended offset shear on the inlet side of one of the primary coolant loops of a four-loop pressurized water reactor (PWR). LOCE L1-5 was conducted in the LOFT facility, which is a heavily instrumented nuclear test system. The system is designed to be representative of large PWRs both in operation and behavior during a loss-of-coolant accident (LOCA). The major components in the LOFT test system include a reactor vessel with a nuclear core, downcomer, and upper and lower plenums; an intact loop, which represents the three unbroken loops of a four-loop PWR; and a broken loop, which represents the broken loop of a PWR. LOCE initiation and size are controlled by two quick-opening valves. The blowdown effluent is contained inside a blowdown suppression tank. ECC injection into the system during a LOCE is provided by an accumulator and high- and low-pressure injection systems. A detailed description of the LOFT system is provided in Reference 5.

The initial conditions for LOCE L1-5 were established at nearly isothermal with the primary coolant temperature at 555 K, pressure at 15.45 MPa, and flow at 176.1 kg/s. The nuclear reactor was in a shutdown condition with the control rods approximately 90% withdrawn. ECC was injected into the intact loop cold leg following the blowdown.

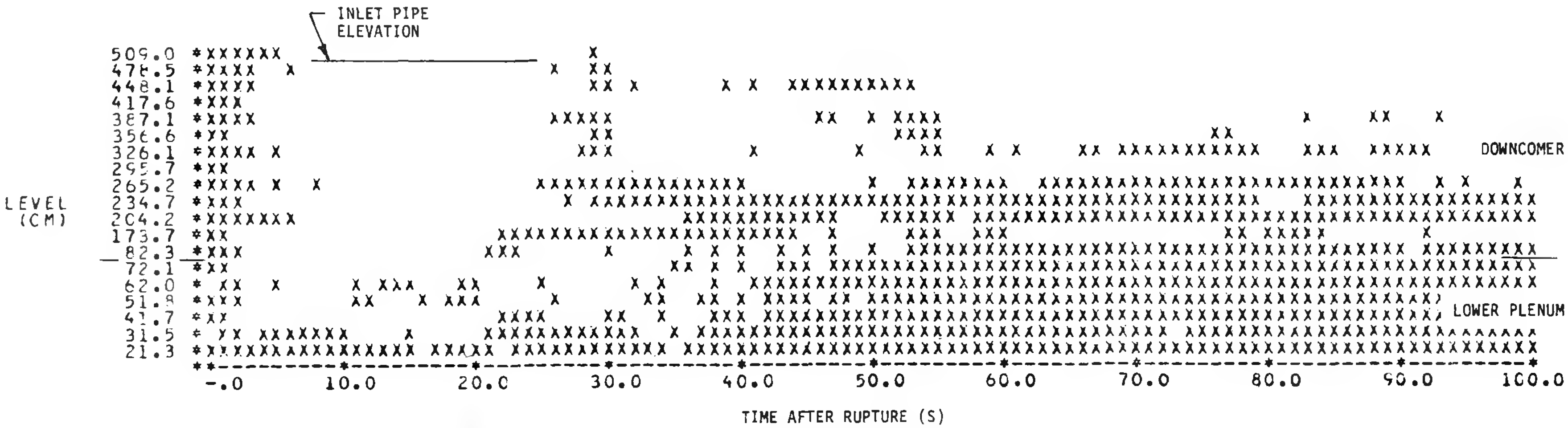
Some of the more significant results from LOCE L1-5 are discussed in the following sections. A detailed data presentation for LOCE L1-5 is provided in Reference 6.

1.1 Downcomer and Lower Plenum Fluid Behavior

Fluid distributions in the reactor vessel downcomer near the intact loop and broken loop cold legs are shown in Figure 10. The blank areas in Figure 10 indicate steam, whereas the symbol "X" area indicates liquid. Figure 10 shows that the downcomer voided in about 3 seconds after rupture, and significant voiding of the reactor vessel lower plenum occurred. The relatively early voiding of the downcomer and the amount of voiding in the lower plenum was different from that observed during LOCE L1-4[7]. A comparison of the results under the intact and the broken loops indicates that more voiding of the lower plenum occurred under the broken loop than under the intact loop. Complete voiding of the lower plenum, however, was not indicated. The lower plenum refilled at about 37 seconds after rupture which was similar to LOCE L1-4. From about 21 to 40 seconds after rupture, a plug of liquid was indicated in the downcomer with some voiding in the lower plenum.



a. Liquid level beneath intact loop cold leg.



b. Liquid level beneath broken loop cold leg.

Fig. 10 Liquid level beneath intact and broken loops in LOFT reactor vessel lower plenum (00.0- to 73.0-cm elevations) and downcomer (73.0- to 485.0-cm elevations).

This phenomena may indicate that countercurrent flow was occurring in the lower section of the downcomer.

Another indication of downcomer hydraulic behavior was obtained from two drag discs located on the instrument stalks in the downcomer near the intact and broken loop cold legs. The results from these instruments are indicated in Figures 11 and 12 where positive flow indicates flow down the downcomer and negative flow indicates flow up the downcomer. A comparison of the results from the drag discs indicates that after ECC injection, at about 19 seconds after rupture, downflow was predominant near the intact loop and upflow was predominant near the broken loop. The results show that asymmetric flow was occurring in the downcomer during the period of ECC injection. This flow behavior was similar to that observed during LOCE L1-4.

1.2 Broken Loop Cold Leg ECC Bypass

An indication that some ECC was bypassing the core and flowing out the broken loop cold leg can be seen in the average density data taken in the broken loop cold leg. Similar ECC bypass was observed during LOCE L1-4 which is compared with the LOCE L1-5 data in Figure 13.

1.3 Core Thermal Response

The core thermal response was obtained from thermocouples attached to fuel rod cladding surfaces. The responses of four thermocouples located 0.203, 0.660, 1.041, and 1.473 m above the bottom of the core on a fuel rod in the center fuel module are shown in Figure 14. The fuel rod cladding temperatures followed the system saturation temperature until about 37 seconds after rupture, when a rapid temperature increase of about 20 K occurred. At other locations in the core, the fuel rod cladding temperature increased at about 27 seconds after rupture. These temperature increases appear to be the result of low steam flow in the core. The occurrence of these temperature increases at different times, depending on the radial locations, may indicate asymmetric flow behavior during core refill. As expected, a rapid reflood of the core from the

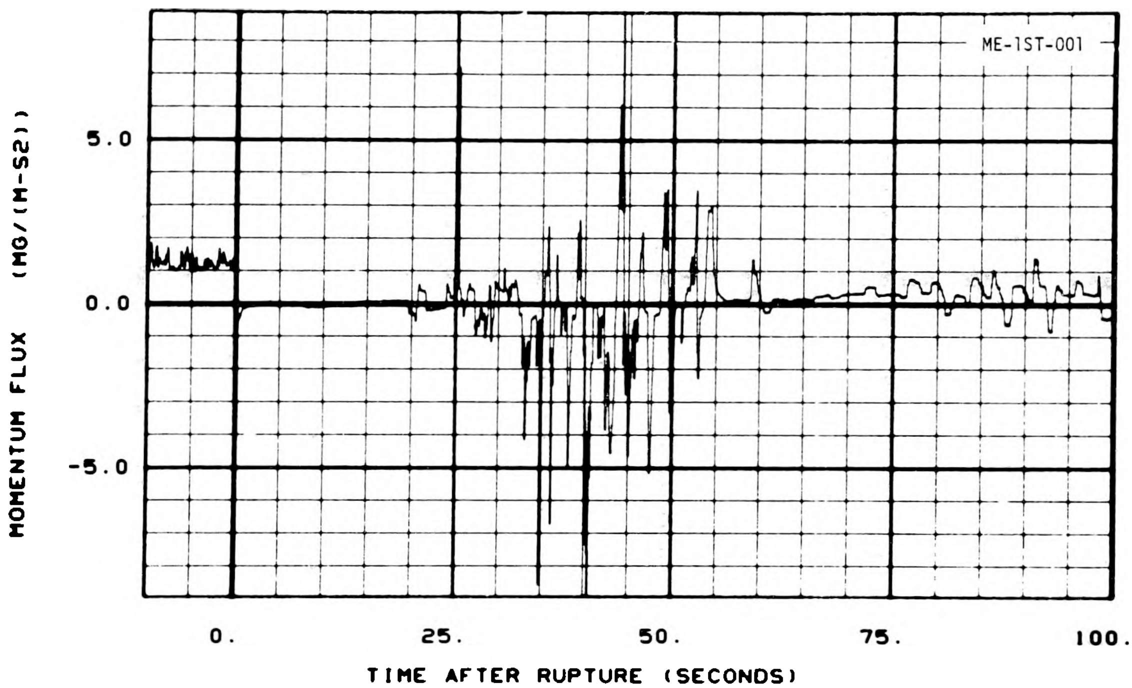


Fig. 11 Momentum flux in LOFT reactor vessel downcomer beneath broken loop during LOCE L1-5.

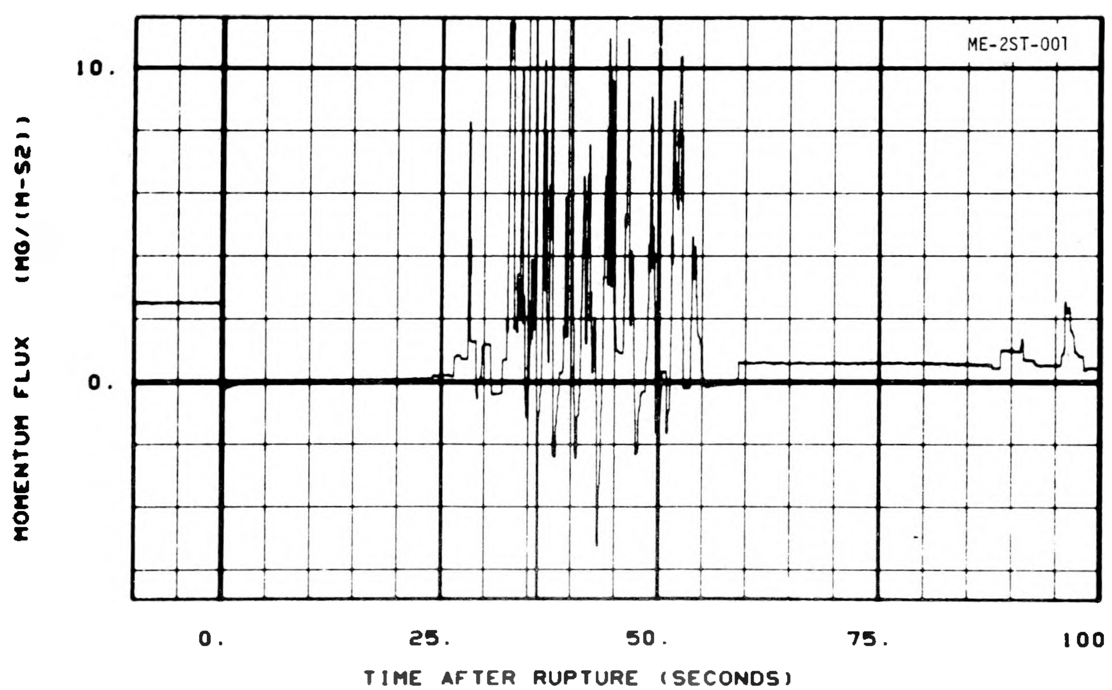


Fig. 12 Momentum flux in LOFT reactor vessel downcomer beneath intact loop during LOCE L1-5.

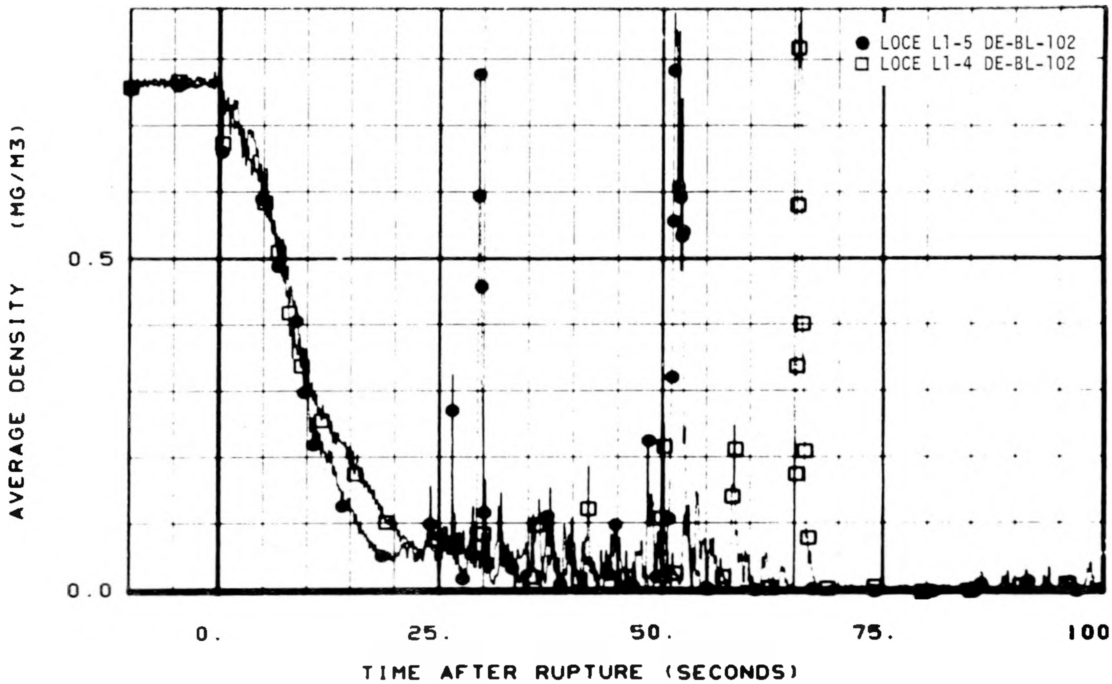


Fig. 13 Average density in broken loop cold leg during LOCEs L1-4 and L1-5.

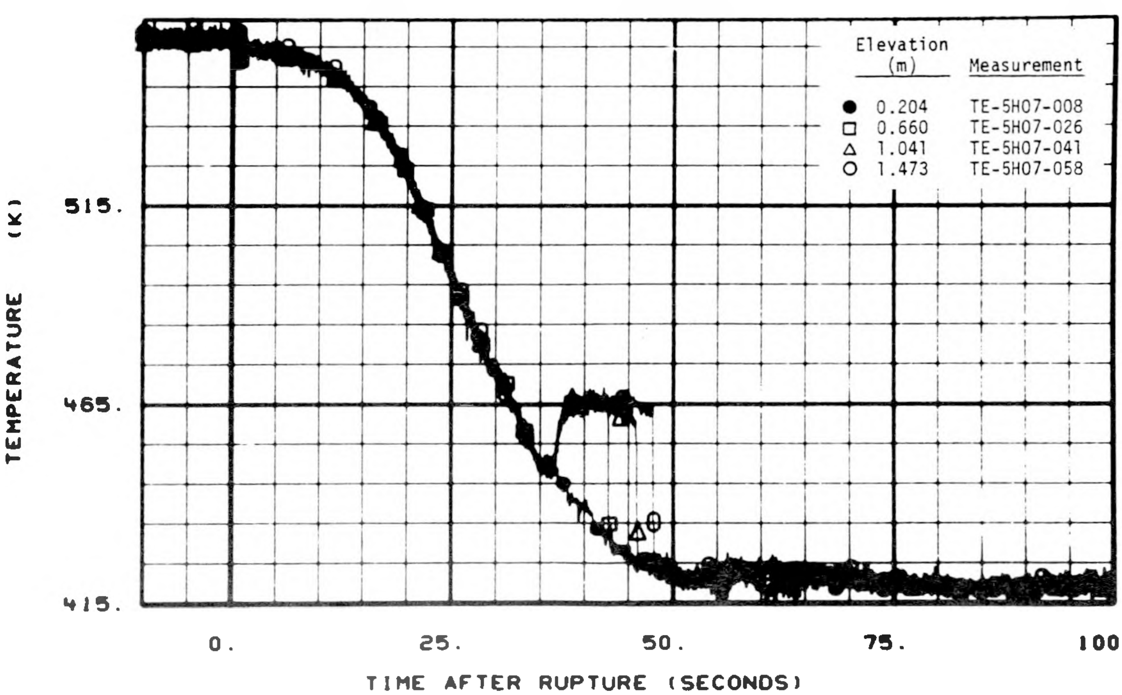


Fig. 14 Axial cladding temperature distribution on fuel rod 5H7 in the center fuel assembly during LOCE L1-5.

core bottom occurred, and the core was filled by about 52 seconds after rupture.

1.4 Conclusions

Conclusions reached after a preliminary analysis of the LOCE L1-5 data are summarized as follows:

(1) Relatively early voiding of the downcomer occurred with partial voiding of the lower plenum.

(2) Asymmetric flow occurred in the downcomer during the period of ECC injection.

(3) Some ECC bypassed the core and flowed out the broken loop cold leg. The amount of bypass was similar to that indicated in LOCE L1-4.

(4) The core flow was asymmetric during refill of the lower plenum.

(5) The fuel rod cladding temperatures increased due to low core steam flow during refill of the lower plenum.

2. LOFT TEST SERIES L2 (POWER ASCENSION SERIES)

H. J. Welland and T. K. Samuels

Test Series L2, scheduled to be started in late 1978, is the first series of LOCEs to be conducted in LOFT with the nuclear core generating heat. Previous LOCEs (Test Series L1) were conducted with a core simulator installed, except for LOCE L1-5 (discussed in Section 1), for which the nuclear core was installed but was not used to produce heat.

2.1 Test Series L2 Objectives

The primary objectives of Test Series L2 are to:

(1) Run the full-power double-ended cold leg break experiment as quickly as possible consistent with an orderly approach to the full-power experiment

(2) Use sequence of events which is representative of best estimate calculations with various 10 CFR, Part 50, Appendix K[8] hardware conditions

(3) Provide data on thermal-hydraulic and fuel behavior and use the data to evaluate and verify computer models and codes that are used to predict large PWR LOCA response

(4) Provide a comparison of nuclear and nonnuclear experimental data to determine and separate the effects of hydraulic forces (and vessel heat transfer) and nuclear heat on the LOCE.

2.2 Test Series L2 Parameters

Test Series L2 is a series of five LOCEs that will be conducted to examine the response of a nuclear reactor to controlled pipe ruptures in the primary coolant system (PCS) cold leg piping. Each LOCE will simulate a 200% double-ended offset shear in the primary coolant cold leg piping. The breaks will be simulated by simultaneous opening of two quick-opening blowdown valves. Table III shows Test Series L2 parameters.

TABLE III

TEST SERIES L2 PARAMETERS

LOCE[a]	Power Level (kW/m)	PCS Flow/ΔT (kg/s)/K	Fuel Prepressurized	Primary Coolant Pumps	ECC Delay
L2-2	26.2	(186.4)/23.8	No	On	No
L2-3	39.4	(181.6)/35.8	No	On	No
L2-4	52.5	(241.9)/35.8	No	On	No
L2-5	39.4	(181.6)/35.8	No	Off	Yes
L2-6	39.4	(181.6)/35.8	Yes	On	No

[a] All LOCEs are 200% double-ended cold leg breaks and all assume loss of one high- and low-pressure injection system train.

The first two tests in the series will be run at intermediate power levels to provide assurance that the full-power test can be conducted safely. LOCE L2-2[a], the first LOCE to be performed in the series, will be conducted at a maximum linear heat generation rate of 26.2 kW/m in the hot fuel rod. LOCE L2-3 will be conducted at a maximum linear heat generation rate of 39.4 kW/m, which corresponds to the power level of present day large PWRs. LOCE L2-4, the full-power LOCE, will be conducted at a maximum linear heat generation rate of 52.5 kW/m, which corresponds to the maximum technical specification limit of typical large PWRs. LOCEs L2-5 and L2-6 will be conducted at the same power level as LOCE L2-3.

[a] LOCE L2-1 has been deleted from the test series. LOCE L2-1, scheduled to be conducted at a maximum linear heat generation rate of 15.8 kW/m, was considered to be conservative.

2.3 Test Series L2 Operating Conditions

Best-estimate conditions require that the LOCE be conducted from initial conditions actually expected in a large PWR should it undergo a LOCA. The basic conditions are (a) no loss of facility electrical power, which requires the primary coolant pumps to operate throughout the blowdown and the emergency core coolant systems (ECCS) to start on signals from the plant protection system, and (b) the nuclear reactor has been generating power for some time. Test Series L2 will simulate these conditions by having the primary coolant pumps operating for 200 seconds after blowdown initiation, starting the ECCS from the plant protection setpoints, and by having the reactor operating long enough to cause a representative decay heat level for the first 200 seconds of the blowdown. LOCE L2-5 will provide data for checking the loss of facility power coincident with LOCA conditions to determine the limiting condition. In LOCE L2-5 the primary coolant pumps will be tripped at blowdown initiation and the ECCSs will be delayed to simulate starting of an emergency power supply. As previously mentioned, LOCE L2-5 will be initiated from the same power level as LOCE L2-3 to provide a direct comparison of the results.

The last LOCE in the Test Series L2, LOCE L2-6, will provide data for assessment of effects of pressurized fuel on the blowdown. Center fuel rods in the center module will be prepressurized to 2.41 MPa. The outer fuel rods of the center module, however, will be unpressurized to allow removal of the module after the LOCE because severe ballooning of the pressurized fuel rods is expected. LOCE L2-6 also will be conducted at the same power level as LOCE L2-3 to provide a direct comparison of the results.

The blowdown effluent will be directed to the blowdown suppression tank. The initial conditions in the suppression tank will be adjusted to give the same initial peak pressure in the tank as is expected in a large PWR containment should the PWR undergo a LOCA. The best-estimate containment pressure for a large PWR is compared with calculated pressure in the LOFT pressure suppression tank in Figure 15. After the peak

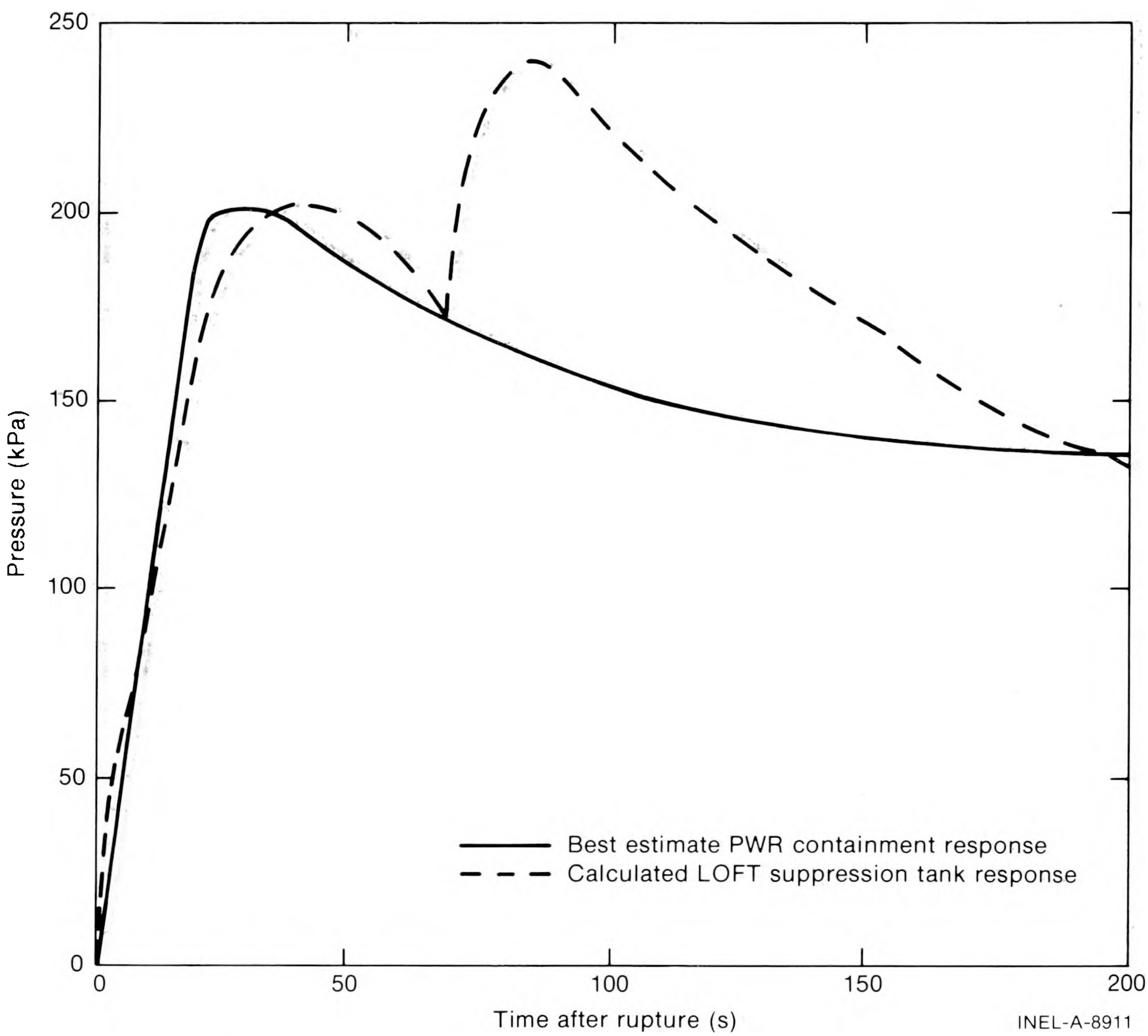


Fig. 15 Comparison of desired (PWR) containment response and predicted LOFT suppression tank response.

pressure in the tank is reached, the blowdown suppression tank spray system will be operated to attempt to effect the same pressure decay in the pressure suppression tank as is expected in the large PWR containment. The LOFT suppression tank spray system, however, cannot control the pressure in the pressure suppression tank after the nitrogen that is used to pressurize the accumulator reaches the tank. The pressure increase caused by the accumulator nitrogen is not expected to have a significant influence on the peak cladding temperature on the LOFT fuel rods.

The ECCS will inject scaled amounts of coolant into the LOFT system during the LOCEs. The LOFT accumulator liquid volume will be volume scaled. The following assumptions will be made:

(1) One of the large PWR accumulator liquid volumes will spill out the break

(2) Volume scaling uses the ratio of the LOFT primary system volume to the large PWR primary system volume.

The calculated LOFT accumulator liquid volume is 1.72 m^3.

The LOFT accumulator gas volume is determined by keeping the ratio of the accumulator gas-to-liquid volume the same in LOFT as in a large PWR. This requires a gas volume of 0.96 m^3.

The low-pressure injection system (LPIS) flow is determined by equating the ratio of the LPIS flow to the flow area of the reactor vessel downcomer, core, and core bypass in LOFT to the same proportion as in a large PWR. The LPIS line is orificed to obtain the curve in Figure 16. Figure 16 also shows the head capacity curve (LPIS unorificed flow) for the LOFT LPIS pump.

The LOFT high-pressure injection system (HPIS) pump flow was volumed scaled, resulting in a calculated HPIS flow of 0.00158 m^3/s.

2.4 Experimental Measurements

Approximately 800 channels of data will be recorded for each LOCE in Test Series L2. Included in these measurements are pressures, temperatures, and differential pressures in the reactor vessel, intact loop, and broken loop; densities at the vessel inlets and outlets;

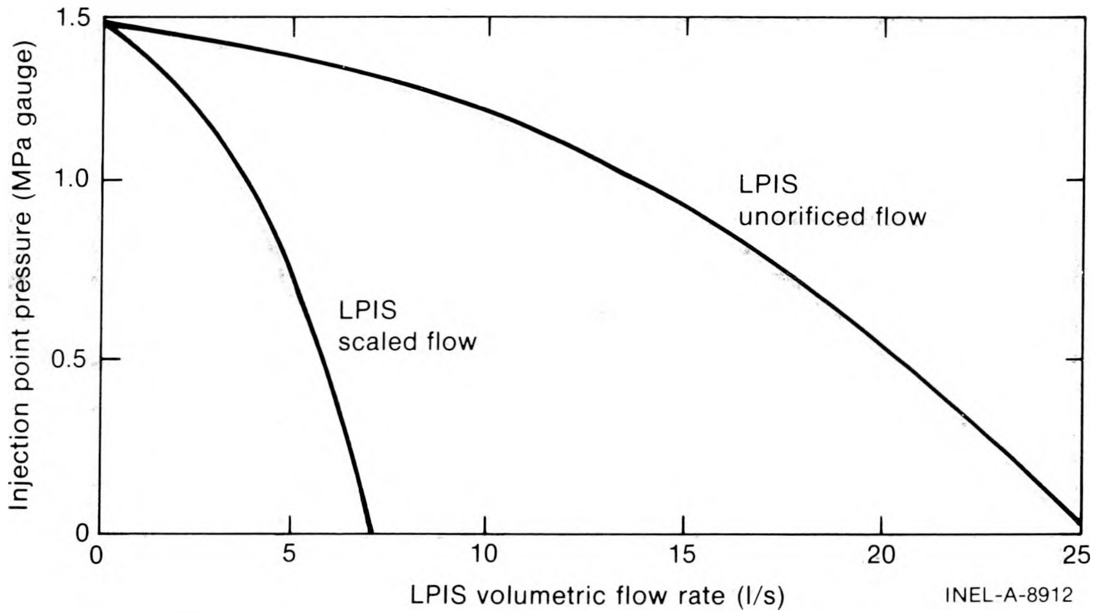


Fig. 16 LPIS flow versus injection point pressure.

liquid levels in the downcomer, core, and upper plenum; and fuel cladding temperatures. Of the 800 data channels, 192 data channels will be measuring fuel rod cladding surface temperatures; 84 of these temperature measurements will be made in the center fuel module.

2.5 Planning Analyses

Planning analyses were performed to calculate the peak fuel rod cladding temperatures that could be expected during Test Series L2 LOCEs. The RELAP4/MOD6[a] and FRAP-T3[b] computer codes were used for these analyses. The peak fuel rod cladding temperatures calculated for the Test Series L2 are shown in Figure 17; results from the analysis are discussed in Reference 2.

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[a] RELAP4/MOD6, Version 3, EG&G Idaho, Inc., Configuration Control Number C0010006.

[b] FRAP-T3: Version 02/16, EG&G Idaho, Inc., Configuration Control Number H00180IB.

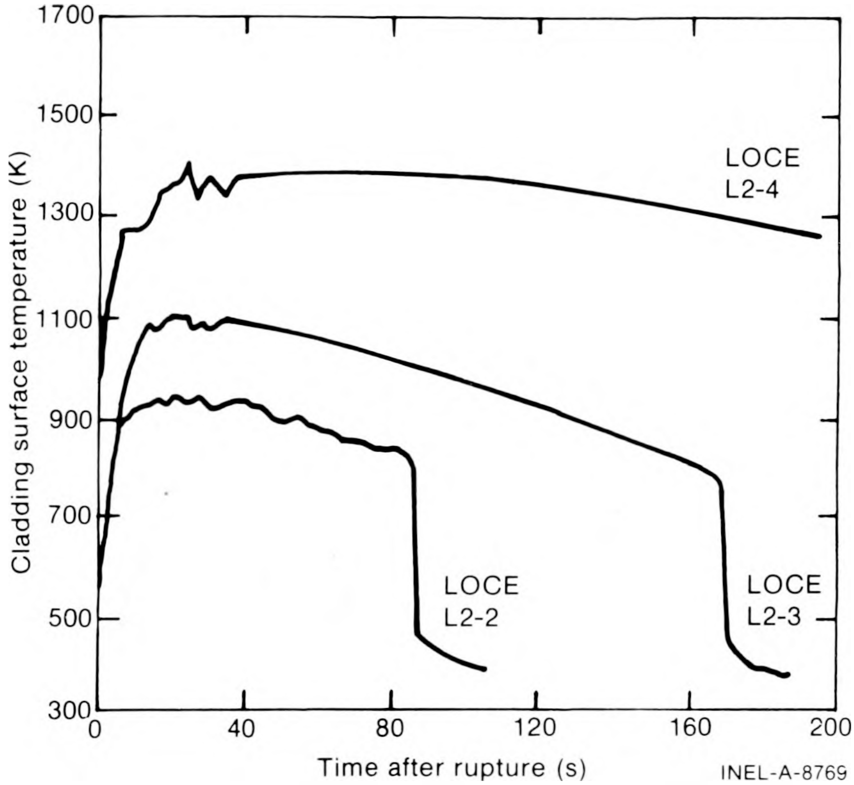

Fig. 17 Predicted peak cladding temperatures for LOFT Test Series L2.

Based on the above analyses, LOFT counterpart tests simulating the Test Series L2 were performed in the Semiscale Mod-1 facility[9]. Data from these tests are presented in References 10 through 14. The peak cladding temperatures observed during the Semiscale Mod-1 tests are shown in Figure 18.

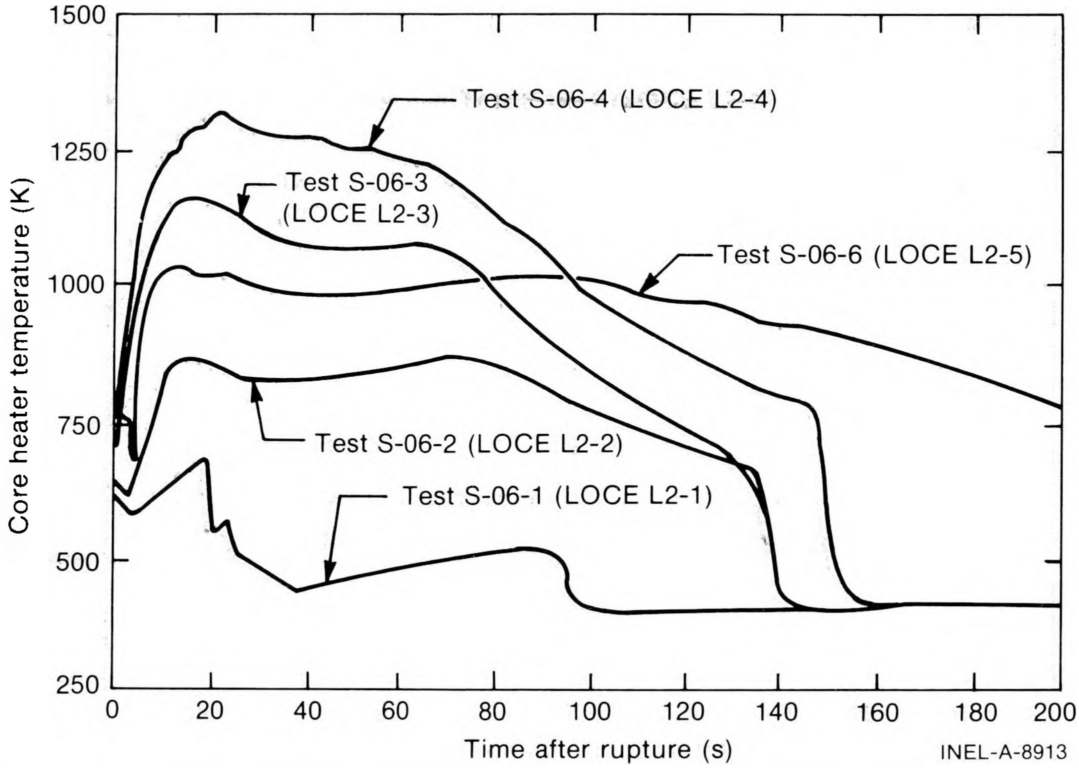

Fig. 18 Peak rod cladding temperatures as measured in Semiscale Mod-1 Test Series 6 (counterpart to LOFT Test Series L2).

III. THERMAL FUELS BEHAVIOR PROGRAM

H. J. Zeile, Manager

J. G. Crocker, Deputy Manager

Accomplishments of the Thermal Fuels Behavior Program (TFBP) have included performance of a power-cooling-mismatch test (Test PCM-5) and five driver core RIA (reactivity initiated accident) lead rod tests in the Power Burst Facility (PBF); preparations for the RIA Scoping Test and Tests RIA 1-1 and RIA 1-2 in the PBF; and reporting of results obtained from tests previously performed in the Power-Cooling-Mismatch (PCM), Loss-of-Coolant Accident (LOCA), and Gap Conductance (GC) Test Series. Work also continued on the Halden Fuel Behavior Research Program and the Power Reactor Postirradiation Examination Program.

The following sections describe (a) PBF testing and (b) activities in the area of program development, data analysis, Halden fuel behavior research, postirradiation examination of commercial power reactor fuel, Nuclear Regulatory Commission technical assistance, and coordination with foreign experimental programs.

1. PBF TESTING

P. E. MacDonald and R. K. McCardell

Tests in the PBF this quarter included (a) PCM-5, the initial power-cooling-mismatch test with a cluster of nine PWR-type rods, and (b) five RIA lead rod tests with a nine-rod cluster of PBF driver core fuel rods in the test space. Results from these tests are being analyzed and will be reported next quarter. Preparations continued for the first programmatic RIA tests (RIA Scoping Test and Tests RIA 1-1 and RIA 1-2). Ongoing activities associated with PBF testing and the analysis of results are summarized in Subsections 1.1 through 1.3. A summary of the results from Test PCM-1 is presented in Subsection 1.4.

1.1 Loss-of-Coolant Accident (LOCA) Test Series

J. M. Broughton, T. R. Yackle, J. W. Spore, and D. R. Evans

During the previous quarter, the first nuclear blowdown tests (Tests LOC-11A, LOC-11B, and LOC-11C) were conducted in the PBF. The peak cladding surface temperatures measured during the LOC-11 tests were slightly lower than desired. Therefore, an extensive experiment hardware design study was performed to evaluate the effects of various potential test train design changes on the expected peak cladding temperatures in Test LOC-3, the next test in the series. This study was conducted using the RELAP4/MOD6 computer code[a]. The design parameters studied were:

(1) Controlled flow bypassing the core (PBF in-pile tube test space)
(2) Downcomer flow area reduction
(3) Upper plenum volume reduction
(4) Check valves in the top of the flow shrouds
(5) Core bypass volume reduction
(6) Flow shrouds extended to particle screen
(7) Flow shroud flow area reduction
(8) Leakage through flow tube and quench line.

All of the design parameters were studied individually and a number of the most promising were studied in combination.

The four design parameters that were found to increase calculated cladding temperature were controlled core bypass flow, (to decrease core

[a] RELAP4/MOD6, Update 4, EG&G Idaho, Inc., Configuration Control Number H00286IB.

flow), downcomer flow area reduction, upper plenum volume reduction, and check valves at the top of the fuel rod flow shrouds. Three principal effects were noted: (a) the duration of the initial negative flow spike was reduced, inducing an earlier critical heat flux (CHF), (b) the core flow was reduced, resulting in poorer post-CHF heat transfer, and (c) quality was increased late in time, also effecting poorer post-CHF heat transfer.

The four design parameters were combined in a single case to ascertain whether their effects were synergistic. Early CHF was assured because the flow shroud check valves severely restricted the initial negative flow spike and the large core bypass flow significantly reduced the total core flow. However, no further reduction in the post-CHF heat transfer was observed in comparison with the single parameter cases. Combining the four design parameters resulted in calculation of higher fluid quality in the core earlier in blowdown than was observed when calculations of the design changes were made separately. This occurred primarily because of the relatively large reduction in coolant mass associated with the downcomer flow area reduction, the upper plenum volume reduction, and the coolant flow reduction. Consequently, rod temperatures late in the transient were higher for the combined case. However, radiation heat transfer from rod to shroud is expected to be important when the quality is high, and this would tend to reduce cladding temperatures. This study was performed with a version of RELAP4/MOD6 which did not include rod-to-shroud radiation heat transfer.

The results of the LOC-3 design study were incorporated into the experiment specifications and hardware design.

After completion of the LOC-3 design study, a radiation heat transfer model was incorporated into RELAP4/MOD6. This modification to the RELAP4 heat transfer package is necessary to accurately predict the cladding thermal response in the single-rod PBF test hardware. The model was debugged and evaluated against the cladding temperature data from Test LOC-11C. A substantial improvement between the calculated and measured cladding temperature reponse was shown for the last 20 seconds of blowdown.

1.2 Reactivity Initiated Accident (RIA) Test Series

Z. R. Martinson, R. S. Semken, T. Inabe, and R. H. Smith

Preparation for the six initial RIA tests continued. Major accomplishments were:

(1) Completion of the RIA Scoping Test experiment predictions including those for reactor physics phenomena (test fuel energy deposition), thermal-hydraulics effects, fuel rod failure mode behavior, and pressure pulse generation. The predictions included the modes of fuel rod failure, energy deposition causing failure, and the consequences of the failures. The failure modes were predicted to be cladding rupture due to high temperature cladding weakening

(2) Completion of a thermal-hydraulic analysis of RIA test fuel rods

(3) Completion and issuance of experiment operating specifications for the RIA Scoping Test

(4) Completion of preliminary experiment operating specifications for Test RIA 1-1

(5) Completion and issuance of experiment specifications for Tests RIA 1-3 and RIA 1-6

The RIA Scoping Test (RIA-ST) analyses indicated that the test fuel rod energy deposition failure threshold for the lower energy phases of the RIA-ST will be 1035 J/g (247 cal/g) at the surface of the fuel. The indicated mode of failure was rupture due to high temperature (near melting) weakening of the cladding. For the final high energy phases of the RIA-ST with radially averaged energy depositions of 1990 and

2510 J/g (475 and 600 cal/g), the mode of failure is calculated to be rupture caused by internal pressurization due to UO_2 vaporization.

1.3 Gap Conductance (GC) Test Series

R. W. Garner and P. H. Klink

Analysis of the data for Tests GC 2-1, GC 2-2, and GC 2-3 was continued. A method was developed for modifying the Ross and Stoute correlation[15] for gap conductance to account for pellet cracking and fuel fragment relocation. The correlation involves a relationship between relocated gap width at hot zero power and the initial as-built gap width, with coefficients determined from a least squares fit of pre- and posttest data obtained from the 12 test rods in Tests GC 2-1, GC 2-2, and GC 2-3. A paper entitled "A Technique for Estimating Relocated Gap Width in Gap Conductance Calculations" was prepared for presentation at the 1978 Winter Meeting of the American Nuclear Society. The postirradiation examination data report for Test GC 2-2 was issued[16].

1.4 Power-Cooling-Mismatch (PCM) Test Series: Results of Test PCM-1

D. T. Sparks and C. J. Stanley

The objective of the PCM test series is to determine the behavior of pressurized water reactor (PWR) type fuel rods operated under various degrees of power and coolant imbalance in film boiling. The specific objective of Test PCM-1 was to extend the present data base from previous single-rod PCM tests to include fuel rod failure at power with sufficient molten fuel present within the rod to produce a high probability of molten fuel-coolant interaction (MFCI) when failure occurs.

1.4.1 Test Design. Test PCM-1 was conducted with a single test rod, typical of 15 x 15 type PWR fuel rods except for overall length and enrichment. The fuel pellets were enriched to 20 wt% U-235 and clad in unirradiated zircaloy-4 to an active fuel height of 0.914 m. The rod was backfilled to a pressure of 2.58 MPa with a mixture of helium and argon gas to simulate an end-of-life gas composition. The fuel rod was positioned in a 16.3-mm inside diameter coolant flow shroud and situated in the PBF in-pile tube. A cutaway schematic of the Test PCM-1 fuel rod and test train configuration is shown in Figure 19.

The test assembly was instrumented to provide a thermal hydraulic energy balance as well as local linear heat rating. The fuel rod was instrumented for measurement of cladding surface temperature, rod internal plenum pressure, and cladding elongation. Pressure sensors in the coolant were provided to detect system pressure and any oversystem pressure pulses.

1.4.2 Test Conduct. Nuclear test operation included (a) a power calibration period during which the rod power with respect to self-powered neutron detector (SPND) current was determined; (b) a preconditioning period which caused cracking of the fuel pellets, accumulation of a fission product inventory, and aging of the cladding surface; and (c) post-CHF operation, during which the fuel rod was operated in stable film boiling until failure, and for approximately seven minutes at full power following failure.

Departure from nucleate boiling (DNB) was initiated by rapidly increasing the test rod peak power from 40 kW/m to 78.7 kW/m at a rate of 33 kW/m per minute while holding the coolant mass flux constant at 1356 $kg/s \cdot m^2$. Coolant inlet temperature and pressure were 600 K and 15.4 MPa, respectively, during the transient. The test rod power was maintained at 78.7 kW/m for approximately seven minutes beyond the time at which rod failure was indicated. The reactor was then shut down and the coolant flow rate was increased to cool the test rod.

1.4.3 Experiment Results and Comparison with Predictions. Fuel rod failure was first detected by a remote area monitor approximately eight minutes after initiation of the power transient. A strip chart recorder monitored the signal from the area monitor. Rod failure was indicated by a step increase in activity from normal background to full scale on the strip chart. Failure was confirmed about one minute later

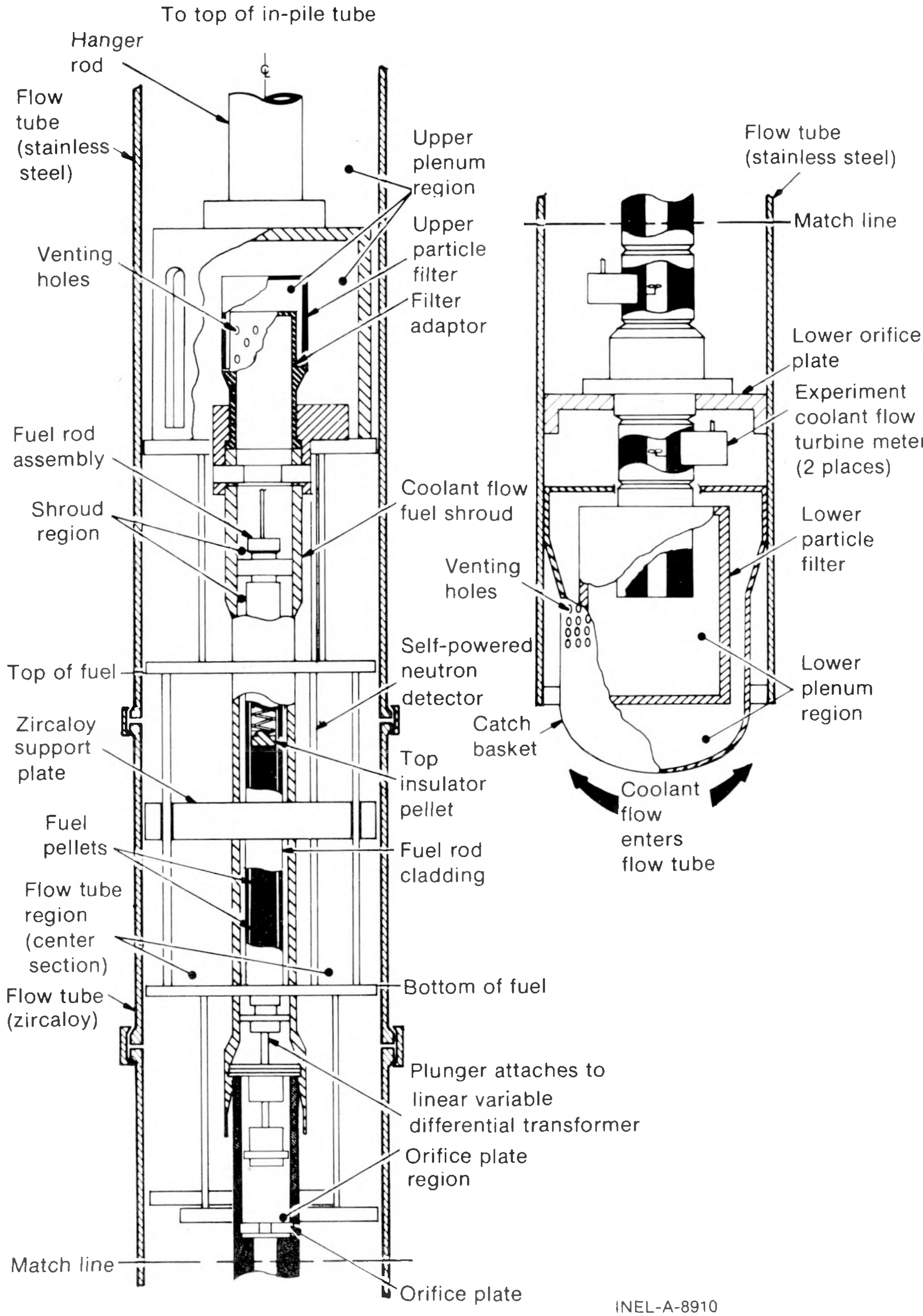

Fig. 19 Schematic representation of Test PCM-1 fuel rod and test train assembly.

by the fission product detection system. Gamma levels in the coolant samples monitored by the detection system increased by two orders of magnitude. Thirty to sixty seconds prior to the detection of fission products in the coolant, the fuel rod elongation instrument recorded a step increase in length of about 3 mm. Rod failure coincided with the change in rod length. The delay time between rod failure and the remote area monitor detection of the failure characterizes the time necessary for the coolant to move from the test train, through the piping, to the cubicle where the monitor is located.

Failure probability calculations for the Test PCM-1 rod were made using FRAP-T3[a] and the modified BUILD5[b] code. FRAP-T3 calculates a failure probability of 55% as the rod enters film boiling with cladding temperature at the hot spot (0.41-m elevation) of about 1700 K. The FRAP-T3 failure probability remains constant at 55% until the equivalent cladding oxidation exceeds 17% of the original wall thickness. Once this criterion is met the code calculates a failure probability of 100%. Calculations of the reaction layer thickness made using the modified BUILD5 program are shown in Figure 20 plotted against time in film boiling for the test rod. Using the reaction layers calculated by the modified BUILD5 code and the 17% equivalent cladding oxidation criterion, a rod failure probability of 100% is reached at approximately 5.5 minutes following film boiling initiation.

An oxygen-stabilized alpha layer will also form on the cladding inside surface due to the diffusion of oxygen from the fuel. Based on previous work[19], the thickness of the inside layer is expected to be about the same as the thickness of the outside layer. At the time of

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[a] FRAP-T3, Version 08/26, EG&G Idaho, Inc., Configuration Control Number H00275IB.

[b] The BUILD5 computer code was developed by R. Pawel at the Oak Ridge National Laboratory, and is based on the mathematical analyses of oxygen diffusion in beta zircaloy[17,18]. The modified BUILD5 computer code listing is presented in Appendix F, Table F-XIV, of Reference 19.

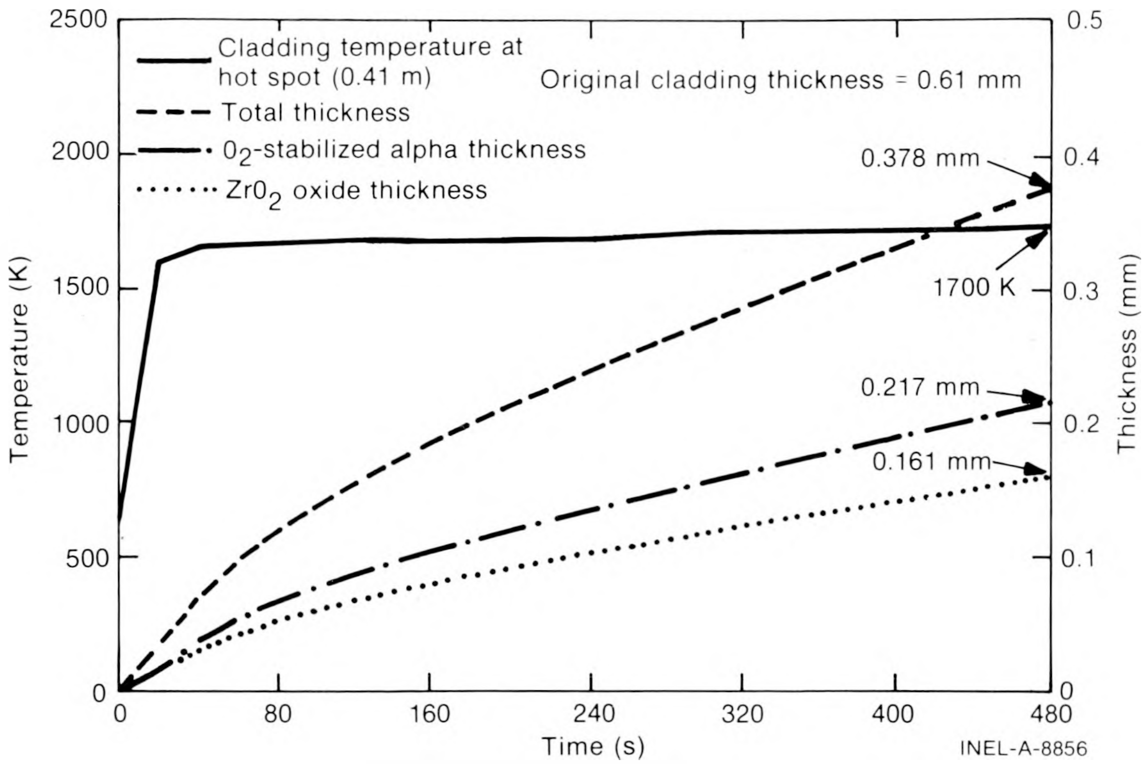

Fig. 20 Modified BUILD5 calculations of reaction layer thickness versus time in film boiling for Test PCM-1 fuel rod.

failure, it appears that nearly all of the zircaloy cladding was converted to oxygen-stabilized alpha or zirconium oxide (0.161 mm of ZrO_2 plus 0.217 mm of oxygen-stabilized alpha on the inside and the outside equals 0.595 mm of reaction product).

Cladding surface temperature measurements and the corresponding FRAP-T3 calculations are shown in Figures 21, 22, and 23 for the three operational thermocouples; the FRAP-T3 calculations end when rod failure is calculated to occur (on the basis of cladding oxidation exceeding 17% of the initial thickness). Measured temperatures of 1360, 1220, and 1050 K at the 0.58-, 0.68-, and 0.78-m elevations, respectively, are in good agreement with the FRAP-T3 calculations at these elevations when the transducers are corrected for cooling fin effects. Cooling fin effects are due to the thermocouple protruding into the coolant channel, resulting in measured readings as much as 200 K less than the actual cladding temperature[20].

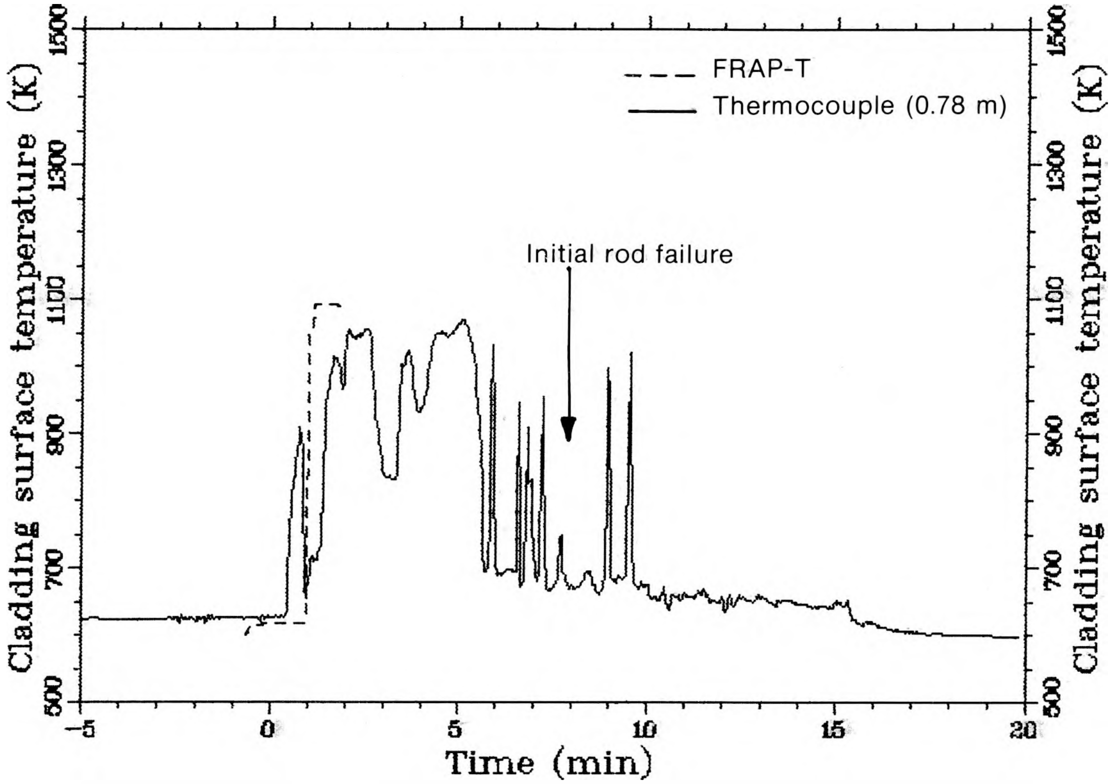

Fig. 21 Comparison of Test PCM-1 with pretest prediction of cladding temperature at 0.78 m from bottom of fuel stack.

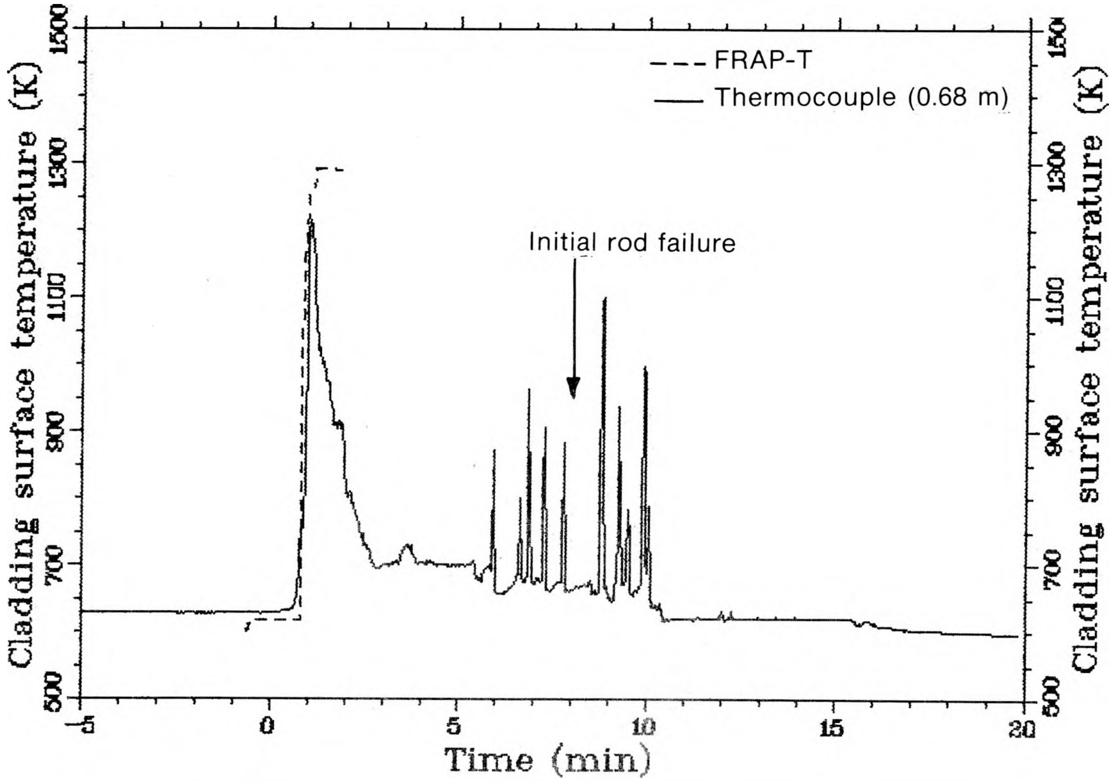

Fig. 22 Comparison of Test PCM-1 data with pretest prediction of cladding temperature at 0.68 m from bottom of fuel stack.

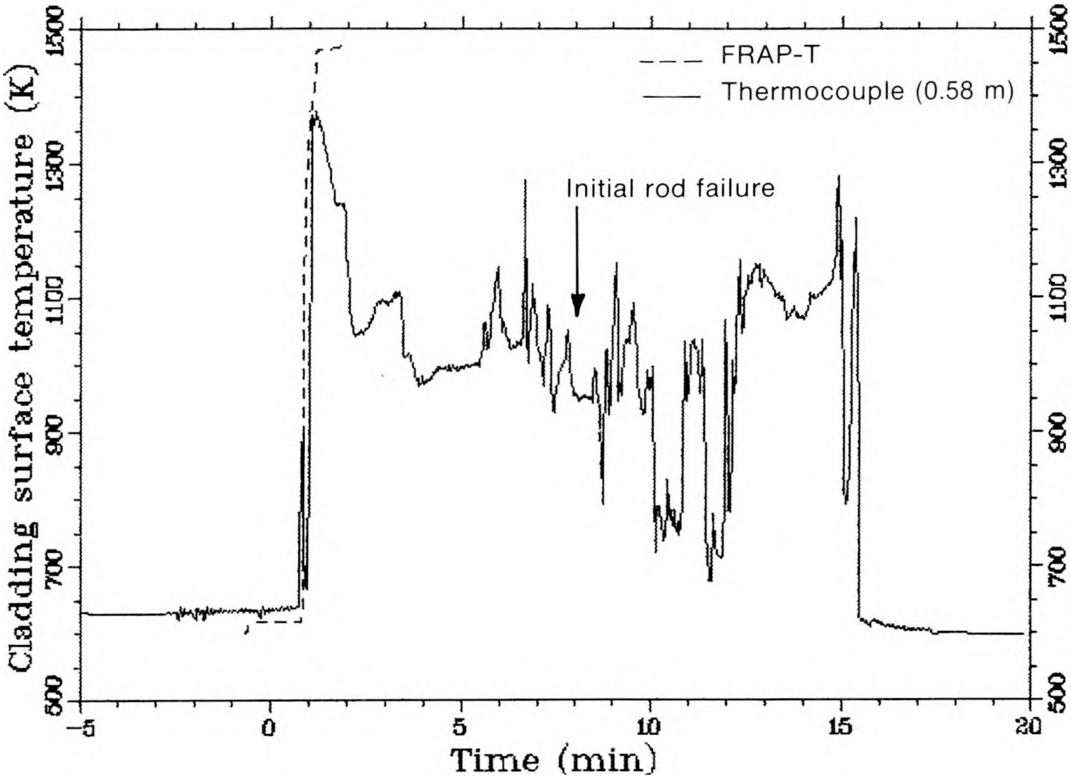

Fig. 23 Comparison of Test PCM-1 data with pretest prediction of cladding temperature at 0.58 m from bottom of fuel stack.

All three operating thermocouples indicated temperatures that decreased by 400 to 500 K during the first five minutes of operation following DNB. The measured temperature profiles indicate a possible change from high temperature film boiling to erratic traces associated with transition boiling. However, the mechanism of this temperature decrease is probably due to oxide buildup under the cladding thermocouples and subsequent breakaway of the oxide, especially at the 0.58- and 0.68-m thermocouple locations. The cladding thermocouple responses shown in Figures 21, 22, and 23 indicate that the temperature decrease occurs earlier in time in the hotter regions of the rod (Figures 22 and 23), supporting the contention that the phenomenon is a function of the oxide buildup. Posttest visual examination of the thermocouple locations also support this theory.

Figure 24 shows the post-DNB history of the fuel rod plenum pressure transducer. At time zero (DNB) the rod pressure rose 2.4 MPa in

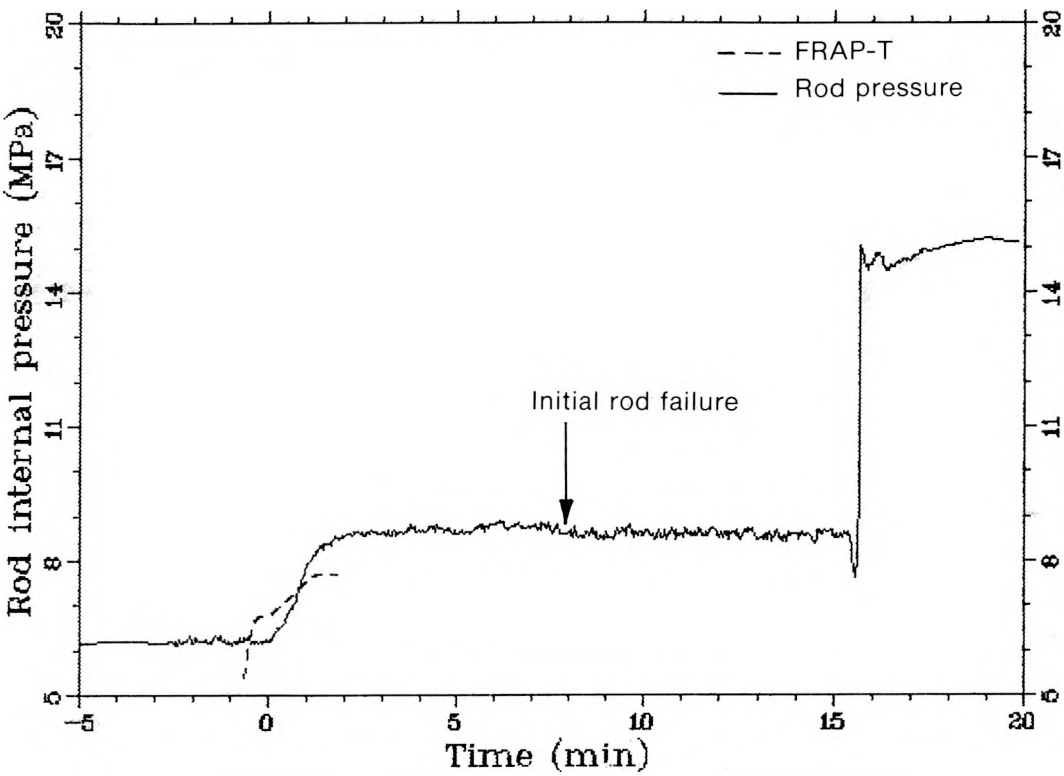

Fig. 24 Comparison of Test PCM-1 data with pretest prediction of fuel rod internal pressure (transducer offset not removed).

response to the higher operating temperature and was in good agreement with the FRAP-T3 calculated pressure increase of 2.3 MPa. The data trace includes an offset which has not been removed.

Rod failure at eight minutes was not detected by the pressure transducer, indicating that the contact between the fuel stack and collapsed cladding was sufficient to seal the plenum area from the failure location.

At 15 minutes after DNB initiation, the test was terminated with a reactor scram. The plenum pressure began to drop as the fuel rod cooled and then increased to the coolant system pressure. As a result of opening the fuel-cladding gap during cooldown, a communication link between the plenum and the failure location was established.

Figure 25 shows the responses from the cladding elongation sensor [linear variable differential transformer (LVDT)], fission product detection system (FPDS), and the internal pressure sensor during the post-DNB test phase. At the onset of DNB, the LVDT indicated a rod length increase of about 6 mm. FRAP-T3 calculations were used to predict a CHF-induced elongation of 7.9 mm, suggesting that rod bowing may have occurred during the temperature transient.

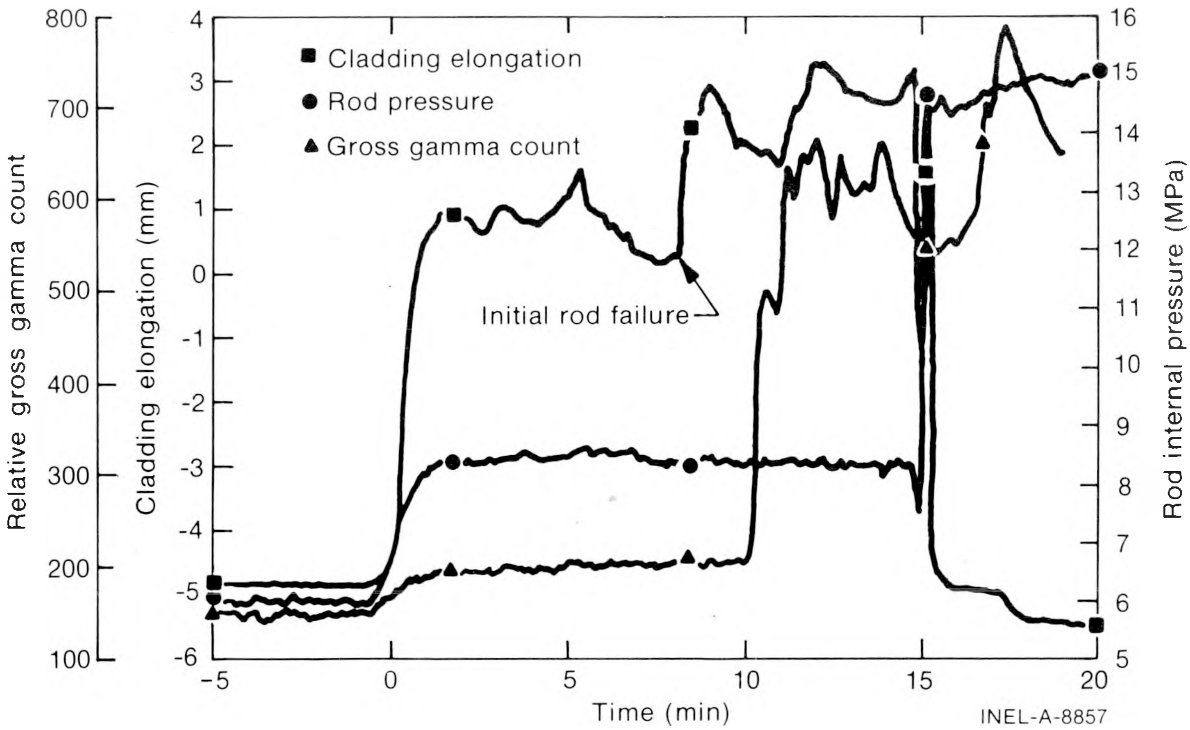

Fig. 25 Test PCM-1 fission product detection system gross gamma count, cladding elongation, and rod internal pressure.

The initial fuel rod failure was indicated by a rapid increase in rod length of 3 mm at about 8 minutes after the onset of DNB. The failure was verified by the FPDS approximately 2 minutes later. The time delay characterizes the time required for the coolant to be transported from the loop to the FPDS gamma measurement device. A second, more massive failure was indicated immediately following shutdown of the reactor (at 15 minutes) by both the LVDT and FPDS. The LVDT showed a rapid increase in length at that time, followed by a large increase in gross gamma counts from the FPDS about 2 minutes later.

The secondary failure was accompanied by a blockage of the coolant flow channel around the fuel rod. Figure 26 shows the flow blockage by comparing the flow into the shroud with the total inlet flow. As the reactor was scrammed (at 15 minutes), the total coolant flow to the test train was increased to cool the fuel rod; however, the flow to the fuel rod decreased to approximately 20% of the prefailure value. The flow blockage is believed to be associated with fuel and cladding debris lodging in the channel between the fuel rod and flow shroud.

Neither of the failures produced the large coolant pressure increase that would have occurred if a violent molten fuel-coolant interaction (vapor explosion) had taken place. A pressure increase of approximately 0.3 MPa was detected following the shutdown failure, but this was believed to be associated with the pump pressure increase necessary to increase the coolant flow rate. Figure 27 shows the pressure increase following the shutdown at 15 minutes. Figure 27 also shows the coolant inlet temperature during the test. Maximum deviation from the nominal value of 600 K is approximately 6 K, indicating very good temperature control during the test.

2. PROGRAM DEVELOPMENT AND EVALUATION

W. J. Quapp and P. E. MacDonald

Activities associated with the PBF program development, coordination with foreign experimental programs, Nuclear Regulatory Commission (NRC) technical assistance, analysis of PBF test results (topical reports), Halden fuel behavior research, and postirradiation examinations of commercial power reactor fuel are summarized in Subsections 2.1 through 2.4.

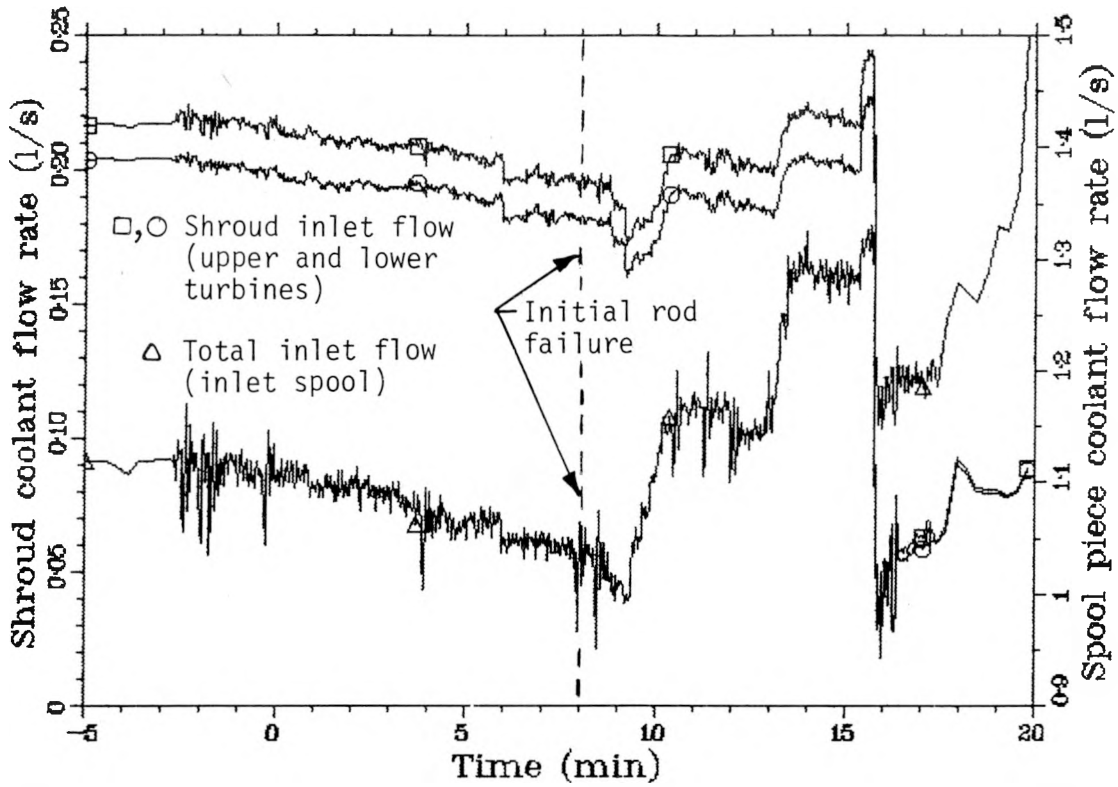

Fig. 26 Coolant flow rates at the shroud inlet and at the inlet spool piece during the DNB phase of Test PCM-1.

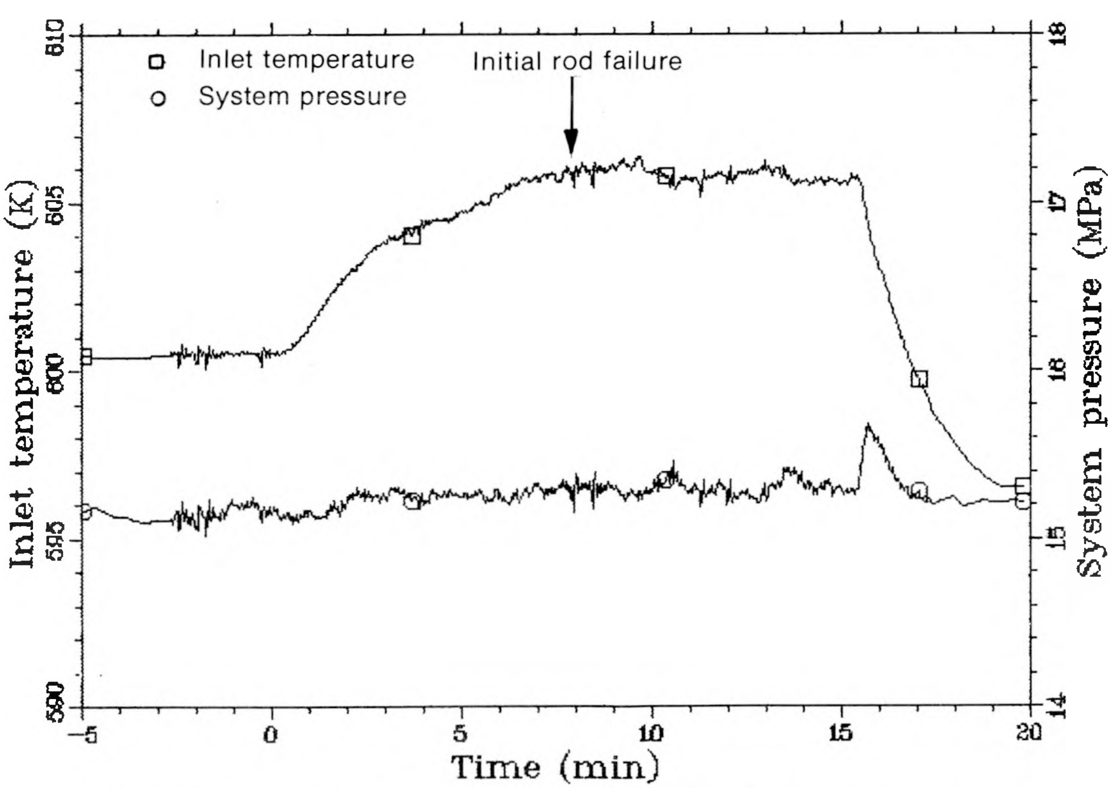

Fig. 27 Test PCM-1 coolant inlet temperature and pressure.

2.1 Program Development and NRC Technical Assistance

S. J. Dagbjartsson and D. W. Croucher

Two staff members participated in the Organization for Economic Cooperation and Development (OECD) Committee on the Safety of Nuclear Installations (CSNI) meeting in Karlsruhe, Germany, to discuss the need for a large-bundle testing program.

An analysis of the power distribution in the seven-rod Halden/NRC reflood experiment designated IFA-511 was completed. The calculations demonstrated a need for greater spacing between the fuel rods to achieve less circumferential power skewing within the rod bundle. A complete report on these calculations is in preparation.

Discussions were held with staff members of the German project for nuclear safety which conducts fuel behavior research for the Federal Republic of Germany regarding the conduct of selected, mutually beneficial tests in the COSIMA facility at Karlsruhe. The tests under consideration will compare the thermal response and deformation characteristics of various types of electrically powered fuel rod simulators under LOCA conditions.

2.2 Topical Reports

D. W. Croucher, A. W. Cronenberg, S. L. Seiffert, and K. Vinjamuri

Results from PBF test programs are being condensed, systematically evaluated, and issued as topical reports covering a number of selected subjects. Key accomplishments during this quarter included (a) completion of a comparison of FRAP-T calculations with experimental data from the PCM experiments, (b) a review of the available vapor explosion theories[21], and (c) an assessment of the performance of previously failed PWR fuel rods during a PCM event[22]. Subsections 2.2.1 and 2.2.2 present a summary of the failed rod behavior and vapor explosion theory assessment.

Analysis and evaluation continued in the following areas:

(1) Review of the SPERT and NSRR reactivity initiated accident test data in terms of fuel rod failure thresholds and mechanisms

(2) Assessment of the embrittlement of zircaloy during a PCM event

(3) Assessment of fuel pellet fragmentation during a PCM event.

2.2.1 Failed Rod Behavior During Hypothesized Accident Conditions. The operation of previously failed fuel rods during hypothesized accident conditions is important to reactor operation and safety. If fuel rods which have experienced a minor loss of cladding integrity can remain in a reactor and operate safely through a hypothesized accident, there may be a substantial economic benefit. Failed rods are now often prematurely removed from a commercial reactor.

Three PWR-type test rods in which failure was detected prior to or very early in their respective tests in PBF were subjected to power ramp and film boiling operation during the PBF power-cooling mismatch tests. The three rods contained a hydride rupture, a pinhole defect, and an axial crack, respectively. These are typical of defects infrequently found in commercial reactor fuel rods.

Rod IE-008 from Test IE-1 failed when hydriding occurred at the peak power location prior to the power ramp. The power ramp which raised the fuel rod peak power to 68 kW/m increased the size of the cladding defect. A fuel-coolant reaction occurred, possibly causing rod bulging near the cladding failure location. Some fuel or fuel-coolant reaction products washed out of the rod through the cladding defects during the power ramp and subsequent steady state operation.

Rod A-0021 from Test PCM-3 is postulated to have had a small manufacturing defect, probably a crack at a cladding thermocouple weld. Postirradiation examination of the cladding disclosed the presence of a

ZrO_2 layer on the cladding inside surface, indicating that water was present inside the rod. The embrittlement of this cladding exceeded that which can be expected from oxidation as a result of cladding-coolant and cladding-fuel reactions and is related to the enhanced pickup of hydrogen in the steam-starved environment of the interior of a fuel rod containing a small cladding defect.

Rod IE-019 from Test IE-5 was initially pressurized to 8.3 MPa to produce cladding failure by ballooning during film boiling operation. After ten seconds of film boiling, a narrow axial crack developed as a result of the ballooning. The rod remained in film boiling for 60 seconds. The rod was severely embrittled during film boiling by oxidation of the interior and exterior of the rod and by the enhanced pickup of hydrogen in the cladding.

Continued operation of the previously failed fuel rods in a power ramp and for a short time in film boiling did not seriously aggravate the condition of the fuel rods beyond that normally associated with the off-normal film boiling operation. Embrittlement in excess of that found in unfailed rods subjected to the same transients was observed in the cladding. The previously failed rods, as well as the unfailed rods, withstood the stresses associated with the quenching of film boiling. Fuel-coolant reaction and fuel washout occurred where large cladding defects were present. However, the molten fuel-coolant interaction produced by extended operation of failed rods in stable film boiling was not observed.

2.2.2 Similarities and Differences in Vapor Explosion Criteria. An overview of recent concepts pertaining to vapor explosion criteria indicates that, in a general sense, a consensus is emerging on the conditions applicable to explosive vaporization. As indicated in Table IV, experimental and theoretical work has led a number of investigators to the formulation of vapor explosion conditions which are quite similar in many respects, although the quantitative details of the model formulation of such conditions are somewhat different.

TABLE IV

COMPARISON OF VAPOR EXPLOSION CONDITIONS FOR VARIOUS MODELS

Vapor Explosion Conditions	Fauske-Henry[23,24]	Board-Hall[25]	Anderson-Armstrong[26]	Cronenberg-Gunnerson[27]	General Model
1. Initially stable film boiling, so that vapor film separates the two liquids and permits coarse premixing without excessive energy transfer	Consistent	Consistent	Consistent	Consistent with all model concepts	Consistent with all model concepts
2. Breakdown of film boiling	Due to thermal or pressure effects	Due to pressure effects	Due to pressure effects	Due to thermal effects	Due to thermal or pressure effects
3. Fuel-coolant contact upon breakdown of film	Liquid-liquid contact	Liquid-liquid contact	Liquid-liquid contact	Liquid-liquid or solid crust-liquid contact	Liquid-liquid or solid crust-liquid contact
4. Rapid vapor production, causing shock-pressurization	Due to spontaneous vapor bubble nucleation (assessed from kinetic theory) and fine-scale fragmentation-intermixing	Due to a large effective heat transfer surface as a result of fine-scale fragmentation and intermixing	Due to a large effective heat transfer surface as a result of fine-scale fragmentation and intermixing	Due to a large effective heat transfer surface as a result of fine-scale fragmentation and intermixing	Large effective heat transfer surface due to frammentation and intermixing; possible, but not necessary, spontaneous nucleation of vapor
5. Adequate physical and inertial constraints to sustain a shock wave	Consistent	Consistent	Consistent	Consistent	Consistent with all model concepts

All vapor explosion model concepts are consistent in that an initial period of stable film boiling, in which molten fuel is separated from coolant, is considered necessary (at least for the large-scale interactions and efficient intermixing), with subsequent breakdown of film boiling due to pressure or thermal effects, followed by intimate fuel-coolant contact and a rapid vaporization process which is sufficient to cause shock pressurization. However, differences arise as to (a) the conditions and energetics that are necessary for film boiling destabilization and (b) the associated mode and energetics of the resultant fragmentation and intermixing. The principal area of difference seems to be the question of what constitutes the requisite condition(s) for rapid vapor production to cause shock pressurization.

To account for such rapid vaporization, Fauske[23] originally proposed that vapor formation occurs at or near the maximum possible nucleation rate, as predicted from kinetic theory. Using Volmer's classical rate equation, a characteristic homogeneous nucleation temperature was assessed. However, simulant fluid experiments[26] have indicated that vapor explosions may occur below such a temperature threshold (which was accounted for in terms of wetting characteristics between fluids), which results in a lower threshold temperature, commonly referred to as the spontaneous nucleation temperature. However, such wetting effect arguments may not account for all simulant fluid experiments where explosive vaporization was observed. In some experiments[26,28] a relatively gradual rise in pressure was noted with thermal conditions rather than threshold events.

A somewhat different concept of rapid vaporization, dating back to early experience with metal-water interactions, is that the phenomena result from fine-scale fragmentation and intermixing of fuel with coolant. The validity of explosive vaporization due to the generation of a large effective heat transfer area, sufficient to cause shock pressurization, has been demonstrated by calculation studies[29,30] and shown to accompany all known vapor explosion events[31]; thus, historically, attempts at understanding the fragmentation process have been a principal area of investigation. Subject to the other conditions in

Table IV, the possibility therefore exists for explosive vaporization by either fine-scale fragmentation and intermixing or by spontaneous vapor nucleation, or a combination of both. Since the UO_2-Na contact temperature is calculated to be well below the homogeneous nucleation temperature, the problem becomes one of assessing the nature and efficiency of the fragmentation and intermixing processes.

As discussed in Reference 32, research efforts with respect to fragmentation have primarily centered on a determination of the principal mechanisms involved. However, to assess the question of whether an MFCI-induced vapor explosion can occur, an understanding of the kinetics of fragmentation, the resultant particle size distribution, intermixing energy considerations, and the heat transfer process between the fuel and coolant must be known; this is not the case at the present time. However, the fact remains that the highest known pressure increase associated with small-scale vapor explosion research occurred when fragmentation and intermixing were initiated in a shock tube[33] or by acoustic means[34]. The results of such experiments provide a strong indication for a vapor film collapse/fragmentation mechanism for explosive vaporization. Thus, Condition 4, as stated in Table IV for what is called a general model, is that fine-scale fragmentation and intermixing are necessary conditions for large-scale vapor explosions; whereas attainment of the spontaneous nucleation temperature need not necessarily be achieved, although it may enhance either rapid vapor production or fragmentation and intermixing.

Although large-scale vapor explosions have been ruled out a priori based on the interface-spontaneous nucleation concept[23], a definitive conclusion that fine-scale fragmentation and intermixing are highly improbable for a reactor environment has not, to date, been demonstrated. Therefore, it appears that explosive vaporization induced by fuel fragmentation and intermixing with coolant should be a principal area of future vapor explosion research. Some primary areas of concern with respect to such fragmentation are:

(1) An understanding of the energetics of vapor collapse for prior film boiling and the effect of vapor collapse on the fragmentation and intermixing energy requirements

(2) An understanding of the kinetics and energetics of fine-scale fragmentation, in the context of the propagating pressure-detonation concept of Board and Hall.

If it can be demonstrated that the fragmentation and intermixing energy requirements cannot be met, then it appears that the potential for the occurrence of a large-scale vapor explosion in a reactor system can be considered negligible.

2.3 Halden Fuel Behavior Research

D.E. Owen

Two papers[35,36] were prepared for the Enlarged Halden Program Group Meeting held in Norway during June. One paper addressed the absorption of helium by the UO_2 in fuel rods initially pressurized to about 2.4 MPa. Based on pressure transducer measurements and postirradiation examination results, only very small amounts were absorbed. The rate of absorption, extrapolated to end-of-life and assuming no fission gas release, would have produced only a decrease of approximately 10% in rod internal pressure.

The second paper dealt with the power distribution within NRC assembly IFA-429. The analysis of thermocouple data and postirradiation fuel rod gamma scans showed that the silver flux shields were only minimally effective in reducing the rod power at the ends of the fuel rods. A three-dimensional Monte Carlo model of the experiment was devised, and calculations made with the model confirmed the experimental results. This model will be used to develop an algorithm to determine the axial power distribution based on the experiment neutron detector signals.

2.4 Postirradiation Examination of Commercial Power Reactor Fuel

D.E. Owen and B.A. Cook

Postirradiation examination of three failed fuel rods from the Peach Bottom 2 boiling water reactor was completed. The rods were from two fuel bundles which apparently experienced pellet-cladding interaction (PCI) failures as a result of an operational power ramp. The fuel rods underwent visual and metallographic examination. Secondary cladding hydriding prevented positive confirmation of the PCI failure mode. Visual examination of the cladding inner surface revealed UO_2 bonding and fission product deposits on both the fuel and cladding. An unidentified white deposit on the cladding inner surface was associated with locally embrittled zircaloy cladding.

IV. CODE DEVELOPMENT AND ANALYSIS PROGRAM

P. North, Manager

The code development effort of the Reactor Behavior Program has been separated from the model verification effort and is now the Code Development and Analysis Program, which has the primary responsibility for the development of codes and analysis methods. The program provides the analytical research aimed at predicting the response of nuclear power reactors under normal, off-normal, and accident conditions. The codes produced in this program also provide a valuable analysis capability for experimental programs such as Semiscale, LOFT, and the Thermal Fuels Behavior Program.

The program comprises the three functional areas of reference code development, fuel analysis research and development, and advanced code development. In the area of reference code development, versions of the reference systems code RELAP4 are developed to allow improved prediction of the thermal-hydraulic behavior of light water reactors (LWRs) during transient or postulated accident conditions. The area of fuel analysis research and development involves developing the FRAP series of fuel rod analysis codes to allow improved analysis of fuel behavior during transient or postulated accident conditions, and is also responsible for improving the materials properties subroutines for MATPRO. Finally, the area of advanced code development is concerned with developing advanced analysis codes for the calculation of the primary system behavior and the containment system behavior during LWR transient and postulated accident conditions; new models and analysis techniques are being developed that have improved capability over those used by existing codes.

During this reporting period, development of RELAP4/MOD7 continued with special emphasis on incorporating an improved best estimate fuel rod model and new core reflood and ECC mixing/refill models. The fuel rod model is based on a subset of the models contained in the FRAP-T fuel rod analysis program. This work is described in the technical contribution in Section 1.2 below.

In the area of advanced code development, work continued on BEACON/MOD2A, an advanced best estimate code for the calculation of the environmental conditions within a containment system during a LOCA, including the thermal and hydraulic effects of wall film and pool formation. Work also continued on the development of RELAP5, a fast-running advanced primary systems code. Preliminary comparisons were made between code predictions and the Semiscale Mod-1 Isothermal Blowdown Test S-01-4A, and good agreement was found.

In the area of fuel rod modeling, programming of FRAPCON-1 was completed. This is the first of a series of improved steady-state fuel rod codes incorporating features from both the FRAP-S codes developed at INEL and the GAPCON codes developed at Pacific Northwest Laboratories. Developmental verification of FRAPCON-1 is underway. New models were added to the FRAP-T transient fuel rod analysis code, including a rod bowing effects model and new fuel rod failure models.

1. LOCA CODE DEVELOPMENT

S. R. Behling, C. H. Burgess, G. W. Johnsen,
R. J. Sand, L. H. Sullivan

Analytical model development efforts have been directed primarily toward the development of RELAP4/MOD7, an integral blowdown/reflood code to be used to analyze large pressurized water reactor (LPWR) behavior.

The principal objectives of the RELAP4/MOD7 development effort are to provide a fast-running, user-convenient code package that includes an integral LOCA analysis capability and offers enhanced modeling capability over previous code versions. An interim version of the code, RELAP4/MOD7, Version 2, to be released in August of 1978, will embody those code additions and revisions generally pertinent to the calculation of an LPWR blowdown. Development activities on Version 2 are essentially complete and developmental verification will commence soon.

Several new models as well as improvements to existing models will be incorporated in RELAP4/MOD7, Version 2. Significant among these and reported here are an automatic self-initialization feature and a linkage to the FRAP-T fuel analysis program. Other code improvements include:

(1) Automatic, decoupled, heat transfer advancement
(2) Improved water property computation
(3) New critical heat flux correlation
(4) New decay heat model and upgraded kinetics model
(5) Two-phase form loss model.

1.1 Self-Initialization Model

An "automatic" thermal and pressure balancing feature has been implemented in the RELAP4 code to simplify the task of initializing LPWR (or similar) models. Heretofore, it has been necessary for the user to compute and input all control volume temperatures and pressures consistent with a steady-state condition. In practice this procedure involves several trial computer runs and tedious hand calculations. The self-initialization package consists of energy balancing and pressure balancing portions.

1.1.1 Energy Balance Model. The energy balancing logic serves to ensure that the total system net heat transfer rate is zero; that is, that the energy input from all sources (core and pumps) is balanced by the removal at all sinks (steam generators). Moreover, the model also ensures that an energy balance exists for each individual control volume.

The user supplies a reference temperature (core inlet fluid) upon which the remaining system temperatures are computed. With the core power and flow rate known, the energy balance model computes the fluid enthalpies and temperatures within each of the intermediate core volumes and at the core outlet. The power contribution of the pump(s) and coolant flow rate determine the temperature rise through the pumps. Deducting this temperature rise from the reference core inlet temperature yields the fluid temperature at the outlet of the steam generator(s). The temperature drop through the primary side of the steam generator(s) is determined by the solution to a set of linearly independent simultaneous equations.

1.1.2 Pressure Balance Model. The pressure balance model ensures that all control volume thermodynamic pressures are consistent with the input relative to flow rate, geometry, and resistance to flow. Present code versions resolve the overspecification of hydraulic input data by introducing a "residual" pressure loss at each junction, which represents the difference between the implied pressure change and that calculated on the basis of mass flow rate, geometry, and form loss coefficient.

The self-initialization pressure balance model minimizes these pressure residuals in all flow loops by adjusting control volume pressures and corresponding loop pump speeds. The user specifies a single control volume reference pressure (e.g., that of the pressurizer) which then represents the location at which the pressure relaxation procedure starts. When the solution is complete, all loops, interconnected or otherwise, possess a pressure distribution consistent with the input geometry, form losses, and flow rates.

When parallel flow paths are encountered by the model (e.g., hot and average core channels) an additional adjustment is required, since the input flow division may not conform precisely to the one implied by the path resistances. In this case, the input form losses for the junctions along the secondary (lesser mass flow) path are adjusted so that the total resistance along the path is consistent with the input flow split.

Dead end flow paths are also accounted for by the model. Along such paths the control volume pressures are computed in accordance with the elevation relationships implied by the input data.

1.1.3 Preliminary Results. The self-initialization model has undergone checkout against RELAP4 models for the Semiscale and LOFT experimental systems as well as an LPWR model. Each model initialized correctly whether the initial input temperatures and pressures were close to or significantly apart from the steady-state values. Figure 28 shows pressures calculated for three control volumes in the LOFT model during a calculation using the transient portion of the code following the self-initialization process. The stability of the pressure levels indicates that a steady-state condition has been achieved.

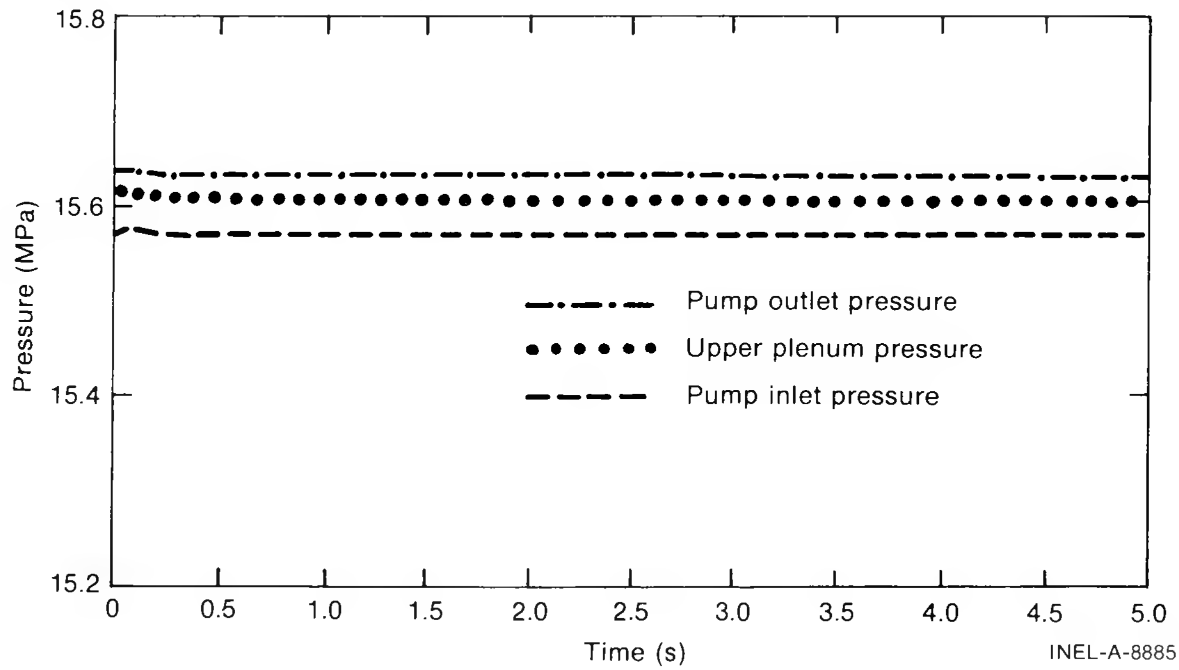

Fig. 28 Self-initialization steady-state LOFT L1-5 model loop pressures.

1.2 RELAP4 Link to FRAP-T Program

A historical shortcoming in the RELAP4 computer code has been the lack of a verified best estimate fuel model. For all earlier versions of the code up to RELAP4/MOD6, representation of the nuclear fuel rod has been accomplished through the use of a simplified cylindrical rod/slab model. This model uses a multiregion (i.e., fuel, gap, cladding) fixed geometry representation of a fuel rod with radial heat conduction, a gap expansion model, and a metal-water reaction model. The importance of thermal and mechanical stress/strain models on stored energy and the prediction of fuel rod integrity led to the inclusion of an abbreviated fuel rod model in RELAP4/MOD6. The fuel rod model was derived from the FRAP-T2 code[37]. Recent advances in fuel model development have rendered this model obsolete, which led in part to the decision to link the RELAP4 and FRAP-T codes together. Such a link offers the potential for quick and efficient upgrading of the fuel model capability within RELAP4, since changes to the mainline FRAP-T program are automatically factored into the RELAP4 program. Moreover, the fuel model in RELAP4 need not undergo future verification beyond initial checkout since the FRAP-T verification as a stand-alone code will be shared by the RELAP4/FRAP-T linked code.

Another advantage resulting from the linked programs is the option to initialize the fuel model with the calculated results from the FRAP-S program via magnetic tape. The FRAP-T portion of the linked codes can read the FRAP-S tape and set the initial condition of the fuel rod in terms of the physical effects of burnup history.

1.2.1 <u>FRAP-S Link to RELAP4</u>. The FRAP-S steady state fuel rod code will be used to provide history dependent initialization conditions to the RELAP4/FRAP-T transient fuel rod model. A multirod core analysis considers fuel rods with varying degrees of in-core exposure. This variation can include, but is not limited to, differences in power history due to the rod's core location as well as accumulated burnup. As a result, the parameters for each of the different fuel rods must be determined prior to a multirod core analysis.

The method used to determine these parameters has been to perform a FRAP-S historical analysis for each of the fuel rods and transfer the appropriate parameters as initial conditions to the RELAP4 fuel model. Reasons for using the FRAP-S code to provide the initial conditions are:

(1) The FRAP-S and RELAP4/FRAP-T analytical and material property models are consistent and compatible

(2) The FRAP-S and FRAP-T initialization link is presently available.

Initialization parameters that are transferred from the FRAP-S analysis to the RELAP4/FRAP-T fuel model, as a function of axial elevation, include: cladding oxide thickness, fuel fission swelling, cladding plastic strain, cladding historical peak temperature, fuel porosity, fuel burnup, plenum gas fractions, internal gas pressure, interfacial pressure, radial temperature profile, and the radial geometry including gap thickness. The foregoing information is transferred to the RELAP4/FRAP-T fuel model at each axial elevation for each fuel rod. This gives maximum flexibility to the transient fuel analysis capability.

Additionally, the capability of initializing the RELAP4/FRAP-T fuel model from card input has been retained. This option is provided in case a FRAP-S analysis of the fuel rod is not available or the user does not wish to generate such an analysis.

1.2.2 Implementation of the RELAP4/FRAP-T Link. The FRAP-T program is capable of independent fuel model analysis; it contains complete models for the solution of heat transfer at the cladding surface (given input hydraulic conditions) and conduction within the fuel rod. RELAP4, as an independent code, also contains these models. Because of the emphasis on thermal-hydraulic phenomena in the RELAP4 code, the heat transfer correlation package is more current, complete, and versatile than that in FRAP-T. Thus the decision to use the RELAP4 heat transfer solution package in the link was straightforward. Whether the conduction solution should be carried out by RELAP4 or FRAP-T was then analyzed. The major consideration leading to the selection of the RELAP4 conduction solution was the complication resulting from combining the reflood moving mesh model in RELAP4 with the conduction solution in FRAP-T. The reflood moving mesh model is designed to accommodate the moving quench front through a dynamic axial nodalization scheme. The model divides the fuel rod into fine, medium, and coarse mesh nodes at each new time point in accordance with the quench front position. Because a fixed length of rod is undergoing continuous renodalization, a sophisticated bookkeeping system is necessary to interface this model with the radial conduction solution. This was already in place in RELAP4/MOD6.

The coupling of FRAP-T with RELAP4 is accomplished by using the outer surface of the cladding as the boundary condition for the RELAP4 heat transfer and hydrodynamic analysis. However, as indicated, RELAP4 performs the conduction solution which produces the radial temperature profile.

In the RELAP4/FRAP-T link, FRAP-T calculates the change in fuel rod diameter, fuel rod internal pressure, and the gap conductance. These values are passed to RELAP4 for each axial elevation of the fuel rod. These properties are then used to solve the RELAP4 heat conduction solution. FRAP-T has been restructured to bypass its heat transfer calculation and to use an axial and radial temperature array constructed by RELAP4. All fuel, cladding, and gap properties computed by FRAP-T are based on the RELAP4-calculated temperature array. Material properties utilized by the FRAP-T code are derived from the MATPRO[38] package. To ensure comparability between a linked RELAP4/FRAP-T calculation and one independently produced by FRAP-T, the thermal properties (i.e., conductivity, heat capacity, and density) utilized by the RELAP4 conduction solution are constructed from the same source.

Previously, in RELAP4/MOD6, a breach in fuel rod cladding integrity was predicated on surpassing a threshold rod-to-coolant channel pressure differential. The failure point pressure differential was input as a function of cladding temperature in tabular form. The RELAP4/FRAP-T linked code utilizes a more sophisticated failure model employed by FRAP-T. The percent of flow channel blockage upon rupture continues to be computed on the basis of the pressure differential existing just prior to rupture.

1.2.3 Preliminary Results Using the RELAP4/FRAP-T Link. Checkout of the RELAP4/FRAP-T link has been performed using a modified form of the six-volume sample problem in the RELAP4/MOD5 User's Manual[39], which is a greatly simplified representation of a 150-MW pressurized water reactor. The checkout was used to verify the link structure and competence of the RELAP4/FRAP-T fuel rod calculation. Since the link had been accomplished in three steps, namely, modification of the FRAP-T code, modification of the RELAP4 code, and linkage of the two parts together, this checkout was performed principally to ensure a proper link and operation of RELAP4 and FRAP-T as an integral unit. Extensive checkout of the link and fuel model capabilities will be performed during developmental verification for RELAP4/MOD7, Version 2.

The six-volume sample problem models a large rupture in the upper plenum region of the reactor. In it, depressurization occurs four seconds after rupture initiation. The fuel rod description was that for 15 x 15 pressurized fuel rods at beginning-of-life and the maximum linear heating rate at steady operation was 6.85 kW/ft. Figure 29 shows a comparison of fuel centerline temperatures calculated by RELAP4/FRAP-T and an independent FRAP-T calculation (the stand-alone FRAP-T calculation utilized the fluid conditions calculated by RELAP4 as boundary conditions). Excellent agreement of these results indicates that the link has been successfully implemented.

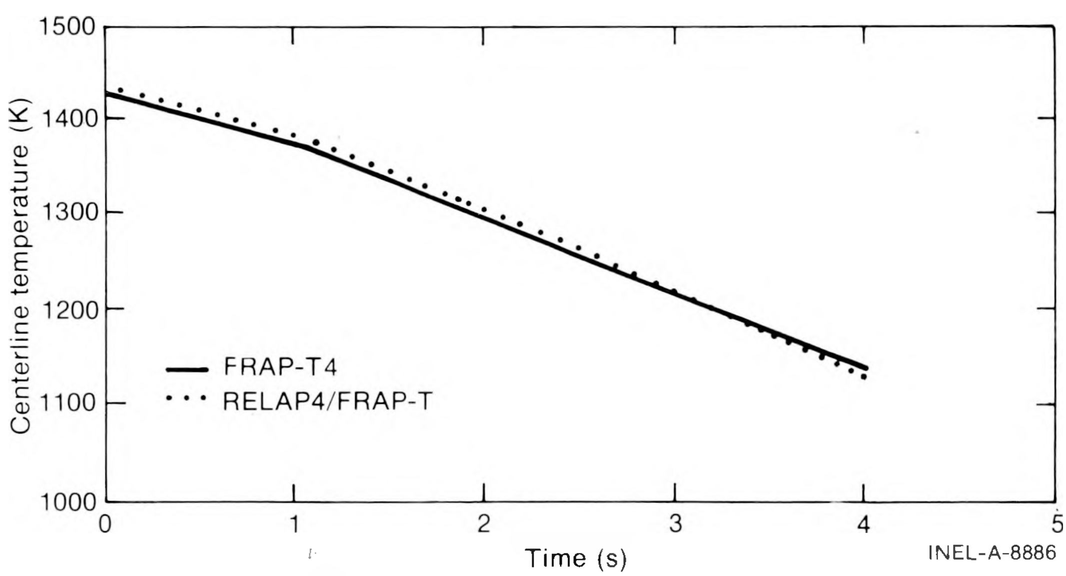

Fig. 29 Fuel centerline temperatures calculated using RELAP4/FRAP-T and using FRAP-T4 for a six-volume sample problem.

V. CODE VERIFICATION AND APPLICATIONS PROGRAM

J. A. Dearien, Manager

The Code Verification and Applications Program is assessing the RELAP4/MOD6 thermal-hydraulic code and the FRAP-T4 fuel behavior code. Of the 18 tasks described in the last quarterly report[2] that are associated with verification of RELAP4/MOD6, 6 have been completed and 10 others have been started or are nearing completion. The FRAP-T4 code/data comparisons and code assessments have been completed and the documentation of the verification is nearing completion.

In support of the U. S. Nuclear Regulatory Commission for industry cooperative safety programs, data comparisons for two tests from the BWR-BD/ECC (boiling water reactor blowdown, emergency core cooling) program were completed. A test prediction for the first FLECHT-SEASET (full length emergency cooling heat transfer-separate effects and systems effects tests) steam generator test was made using RELAP4.

The NRC/RSR data bank has been established and 7.4 megawords of data have been deposited during this quarter. The first version of the data bank processing system (DBPS) has been completed and released to users at INEL.

1. LOCA ANALYSIS VERIFICATION

T. R. Charlton, W. S. Haigh,
T. D. Knight, and S. G. Margolis

Five of the completed RELAP4/MOD6 verification tasks were completed in this period: two blowdown separate effects (i.e., core and pressurizers) tasks, a reflood separate effects task, and two reflood system tasks. In addition, three test predictions and estimate error bounds for calculating peak cladding temperatures have been completed.

As part of the independent verification for RELAP4/MOD6, calculations of RELAP4 core models were compared with blowdown data from Semiscale Test S-06-5 and THTF (thermal-hydraulic test facility) Test 105. Both blowdowns were initiated by a simulated large cold-leg break from typical PWR conditions of 15.6 MPa and 555 K core inlet coolant temperature. Test 105 had a peak power density of 55.6 kW/m, while Test S-06-5 had 39.7 kW/m. Both facilities are electrically heated.

RELAP4 component models, driven with measured boundary conditions, were used to calculate core behavior during the two experiments. Base runs were made consistent with guidelines established by developmental verification.

Heater rod temperature response, both calculated and measured, was similar for the two experiments. In the lower half of each core, CHF (critical heat flux) generally occurred early in the tests (between 0.5 and 1.0 second after rupture). After CHF, measured cladding temperatures increased and eventually peaked. Calculated CHF and cladding temperature response low in the core agreed closely with measurements. Figure 30 shows a comparison of calculated peak cladding temperatures and measured values for low elevations in the cores and includes the high power step of the axial power profile for both tests. The elevations indicated in Figure 30 are the distance above the bottom of the heated lengths of the cores (THTF core length is 3.65 m; Semiscale Mod-1 core length is 1.7 m). Cladding temperatures were well correlated, although significant scatter, caused by variation in the time to CHF, existed in the measured temperature response at the Semiscale peak axial power step. The calculated peak cladding temperature was close to the maximum of the measured values.

In the upper half of each core, CHF generally did not occur before 2 seconds following initiation of the transients. However, RELAP4 continued the trend of predicting early CHF established in the lower core

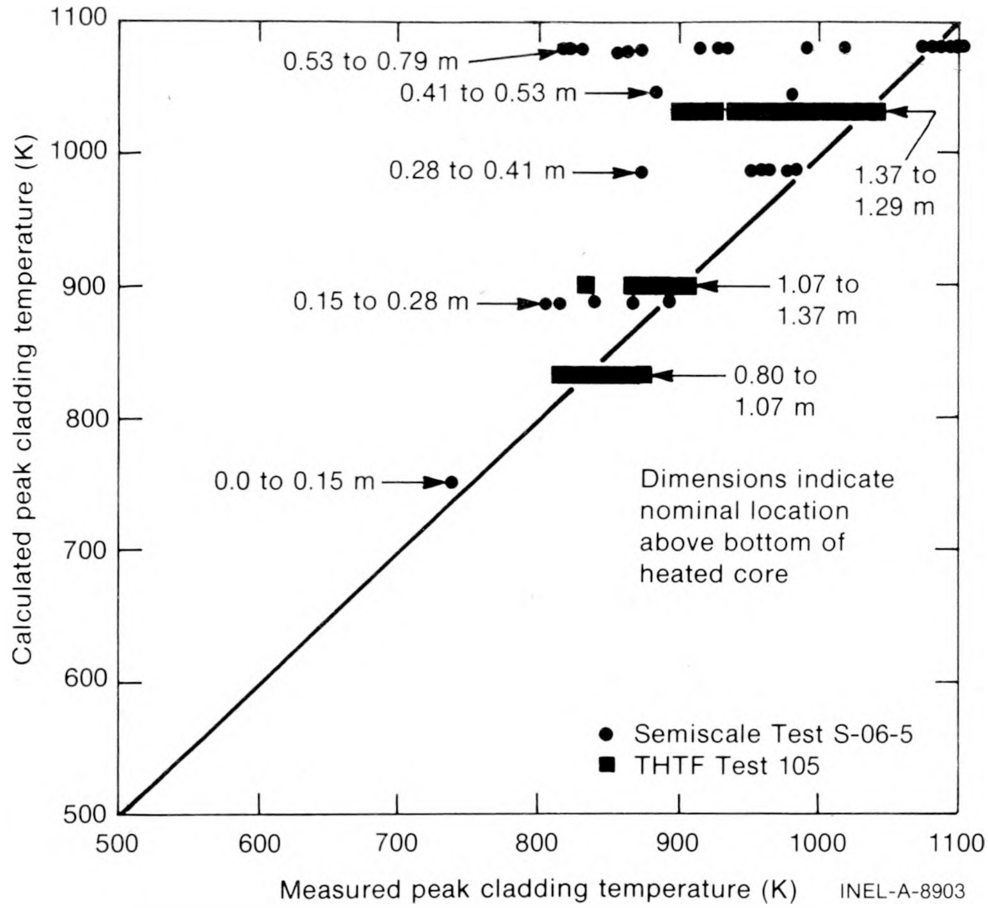

Fig. 30 Comparison of measured and calculated lower core cladding temperatures during blowdown, Semiscale Test S-06-5 and THTF Test 105.

calculations. The prediction of early CHF resulted in overpredictions of peak cladding temperatures by as much as 200 K. Figure 31 compares calculated peak cladding temperatures with measured values for both tests at elevations above the axial peak power step.

In those instances where CHF was calculated accurately in the base runs, cladding temperatures were calculated accurately for the entire blowdown. Where CHF was calculated poorly, cladding temperatures were calculated poorly for the entire blowdown. An improved CHF correlation is needed in RELAP4.

Additional studies were made to determine the sensitivity of the base calculations to CHF correlation, core phase slip model, core model

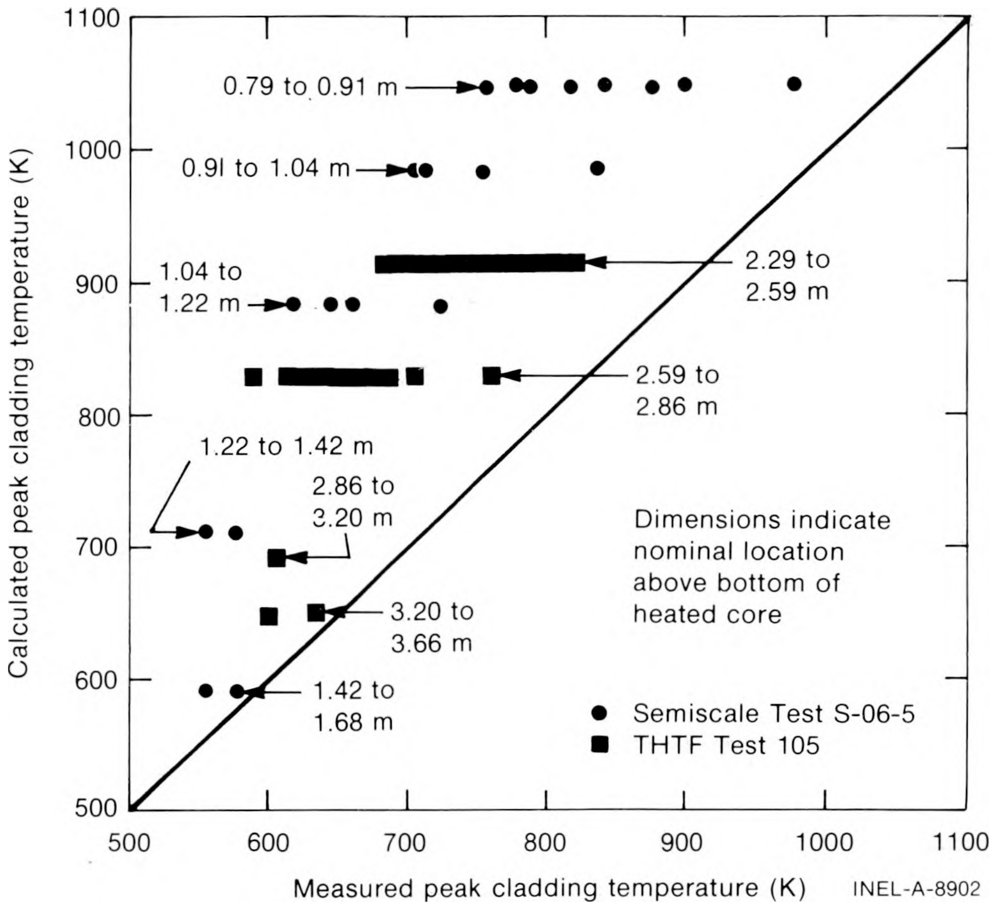

Fig. 31 Comparison of measured and calculated upper core cladding temperatures during blowdown.

nodalization, uncertainty in measured boundary conditions, and film boiling correlations.

A better calculation of CHF high in the core was obtained with the GE (General Electric) CHF correlation than with the recommended RELAP4/MOD6 correlations (which are the W-3, Hsu and Beckner modified W-3, and modified Zuber correlations). However, unrealistic rewets in the lower core were calculated with the GE correlation. Although the calculations indicate the GE CHF correlation may not be fully applicable to best estimate PWR analysis, they did demonstrate the adequacy of the RELAP4/MOD6 blowdown heat transfer correlations (HTS2) with the modified Tong-Young correlation for transition boiling and the Condie-Bengston III film boiling correlation.

Cladding temperatures were calculated more realistically using the Condie-Bengston III film boiling correlation than with either the Groeneveld or Chen nonequilibrium film boiling correlations. Higher cladding temperatures were calculated with the nonequilibrium correlations.

The use of the vertical phase slip model in the core decreased the accuracy (relative to measurements) of calculated core boundary conditions. Because the phase slip model also calculates slip velocities which are probably too large for cores, the vertical slip model generally should not be used to analyze PWR type cores.

Calculated peak cladding temperatures were converged with respect to core nodalization when the cores were modeled with five control volumes. (Most current RELAP4 models of the Semiscale and THTF systems use five control volumes to model the core.) An increase in the number of core volumes did not significantly affect calculated cladding temperatures.

Calculated cladding temperatures were sensitive to variations in the flow boundary conditions of the RELAP4 models. Variations in the flow, which were representative of measurement uncertainty, did not significantly alter the calculation of CHF. Thus, in the base runs, errors in the time to CHF at elevations above the high power step were probably not caused by inaccuracies in the flow measurements used to drive the component models. Flow variations did result in a 50-K variation in calculated peak cladding temperature and significantly affected the calculated rate of temperature decrease after peak. At the end of blowdown, cladding temperatures were insensitive to variations in the pressure boundary conditions.

Three verification studies were made to compare the results of RELAP4/MOD6 calculations with reflood experimental data. One deals with the results of code application to describe experimental behavior in core reflood separate-effects tests; the other two studies present comparisons with selected reflood systems-effects test data.

One test was selected from each of three forced-feed reflood experimental series and was analytically modeled using guidelines set up for overall assessment consistency. The tests selected were:

(1) Westinghouse FLECHT [low flooding rate (LFR) cosine bundle] Test 4019[40]

(2) Westinghouse FLECHT (LFR skewed bundle) Test 11003[41]

(3) Semiscale Mod-1 Test S-03-D[42].

The boundary conditions for each model were taken from the experimental results. Calculated mass response in the core showed inventories generally within ±7.5% of the experimental results for the two FLECHT tests. The code-calculated mixture-level response for Test S-03-D was significantly lower than that of the experiment during the last half of the reflood. For FLECHT Test 4019, the code overpredicted the bundle hot spot (midplane) peak cladding temperature, turnaround time, and quench time by 46 K, 35 seconds, and 22 seconds, respectively. Otherwise, the code generally underpredicted the experimental results at low elevations and overpredicted them at high elevations. For both FLECHT Test 11003 and Semiscale Test S-03-D, the cladding temperature, turnaround time, and quench time were overpredicted at all elevations, indicating that the user guidelines set up for the calculations are not universally applicable.

The second study was made to assess application of RELAP4/MOD6 to two gravity-feed reflood experiments: FLECHT-SET Test 2714B and Semiscale Test S-03-5. Each of these reflood systems experiments was selected because it was well documented and significantly different from previously analyzed tests of the same series. A calculation was performed for each test and pertinent code-calculated hydraulic and thermal results were compared with experimental data. The comparisons indicated good agreement of calculations and data for core mixture level response, but not for rod cladding temperature responses. Causes for unfavorable thermal agreement were investigated, identifying:

(1) A low-temperature dispersed-flow heat transfer anomaly, for which it was recommended that a more appropriate correlation be used in the calculation.

(2) Rapid flucuations in steam-generator heat transfer behavior that greatly affected system thermal and hydraulic responses. A possible method for eliminating these fluctuations was recommended.

(3) Sensitivity to selection of transition and film boiling input options, warranting better definition of the use of these options.

Sample data from the forced-feed and gravity-feed experiments of the FLECHT, FLECHT-SET, and Semiscale Mod-1 programs are compared with code calculations in Figure 32. In Figure 32, the calculated peak cladding temperatures are plotted as a function of experimental measurement data. From these data, an estimated mean error of 95 K and a standard deviation from the mean of 134 K were calculated. Based on this information a prediction interval for the data described in the third report was determined.

The third study presents the results of code/data comparisons based on test predictions of two West German PKL reflood systems experiments, Tests K5A and K7A. The predictions were considered "double blind," because the verifiers had been allowed no access to the data and no prior modeling application of the RELAP4/MOD6 code had ever been made to the PKL facility. Preliminary test predictions had been made prior to release of the data for code evaluation purposes. Data comparisons show that the code calculated maximum cladding temperatures to within 3% for Test K5A and 5% for Test K7A. Deviations were greater in lower and

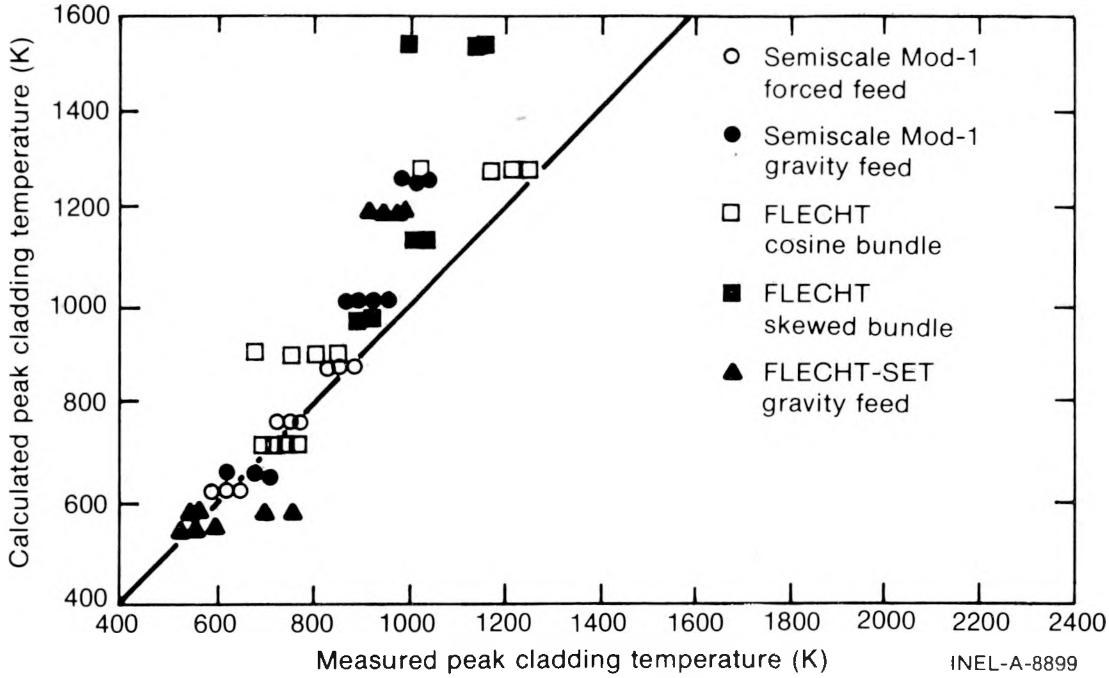

Fig. 32 Comparison of calculated and measured peak cladding temperatures in Semiscale Mod-1, FLECHT, and FLECHT-SET experiments.

upper core regions than in the middle core region. At higher elevations in all three radially defined core energy zones, quench occurred earlier in the experiments than in the calculations. A summary of the hot-channel peak cladding temperature data for PKL Tests K5A and K7A is presented in Figure 33. The calculated versus measured format of Figure 33, is the same as in Figure 32 except that the 50% confidence level prediction interval is also shown. Of the 51 points obtained, only 2 lie outside the prediction interval, which is an indication of the applicability of the code-assessment technique for using the results of code/data comparisons for one set of experiments to calculate the results of another set of different scale.

The hydraulic modeling of the experimental system generally simulated the experimental loop flow histories. However, some core oscillation amplitudes were overpredicted, and a calculational sensitivity to steam-generator gas vaporization dynamics was identified. Areas were identified in which a need exists for further code development and improvement. The most significant of these areas are in core liquid entrainment and dispersed-flow heat-transfer modeling.

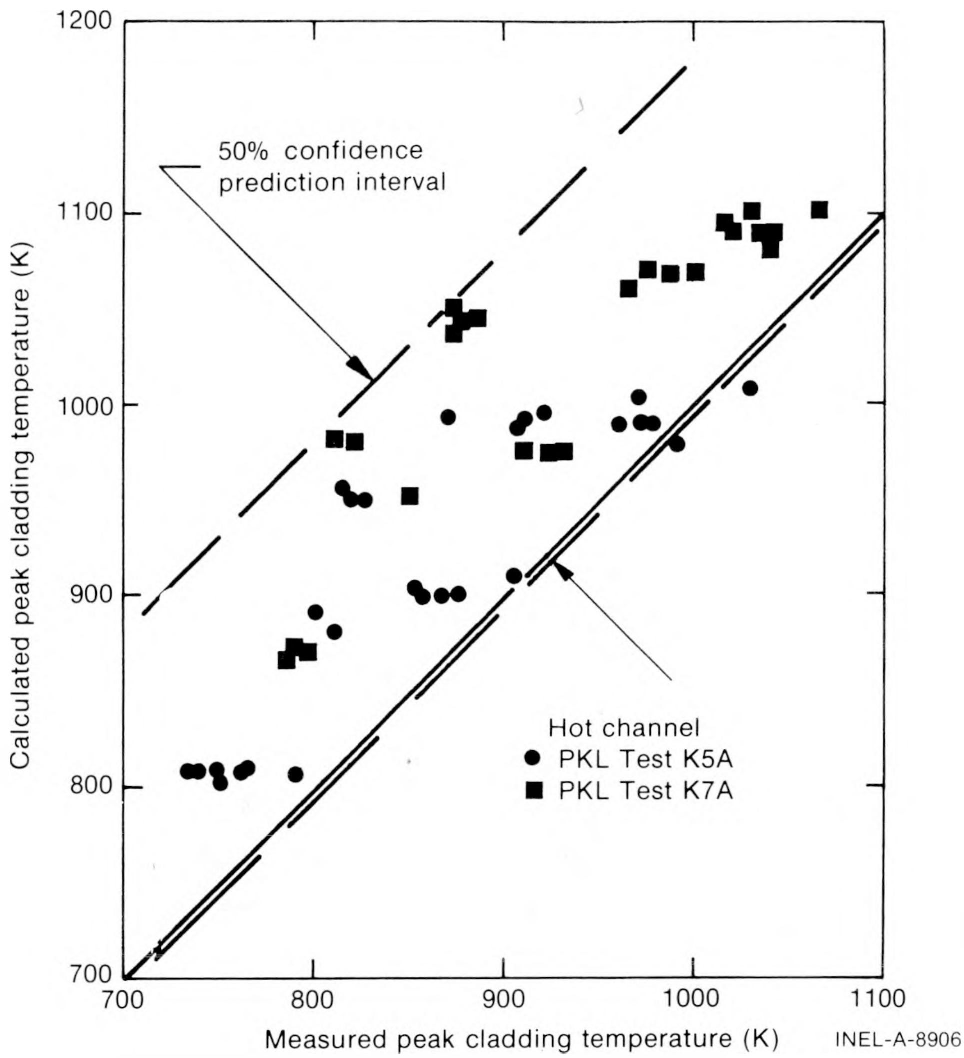

Fig. 33 Comparison of calculated and measured rod surface temperature for PKL tests.

In addition to the code/data comparisons described above, three test predictions were completed. The test predictions were for LOFT Test L1-4 and Semiscale Tests S-07-1 and S-07-4. In calculating the results of Test S-07-1 and Test S-07-4 (the first Semiscale Mod-3 reflood test), an estimate of the uncertainty of RELAP4/MOD6 peak cladding temperature calculations was made using Semiscale Mod-1 test data. The primary differences between the Semiscale Mod-1 system and the Mod-3 system are:

	Semiscale Mod-3	Semiscale Mod-1
Rod length	3.66 m	1.68 m
Axial power profile	Symmetrical	Bottom skewed
Broken loop	Active pump and steam generator	Simulated pump and steam generator
Downcomer geometry	Pipe	Annulus

Use of Mod-1 experience to successfully calculate the range of errors in Semiscale Mod-3 tests will provide evidence that RELAP4/MOD6 is not tuned to the geometry and scale of the Semiscale Mod-1 system. Such predictions have been made and the results are shown in Figures 34 and 35.

Experimental results from Semiscale Mod-3 Test S-07-1 and their relation to the error bounds shown on Figures 34 and 35 will provide one measure of the present effectiveness of best estimate calculations combined with statistically established error bounds.

2. FUEL ANALYSIS VERIFICATION

Fuel model verification efforts were directed toward systematic execution of the many FRAP-T4 runs required to support various analytical and data comparison studies. These studies consist mainly of data-prediction comparisons for key thermal and mechanical fuel behavior parameters measured under quasi-steady state, off-normal, and transient operating conditions. A few steady-state test rod and commercial rod cases are considered to compare initial transient conditions with those calculated by the more extensively characterized normal condition model, FRAP-S3. The inclusion of out-of-pile data comparisons was necessary to evaluate high temperature cladding rupture models because of the relative lack of operating rod data under blowdown conditions. In all, some

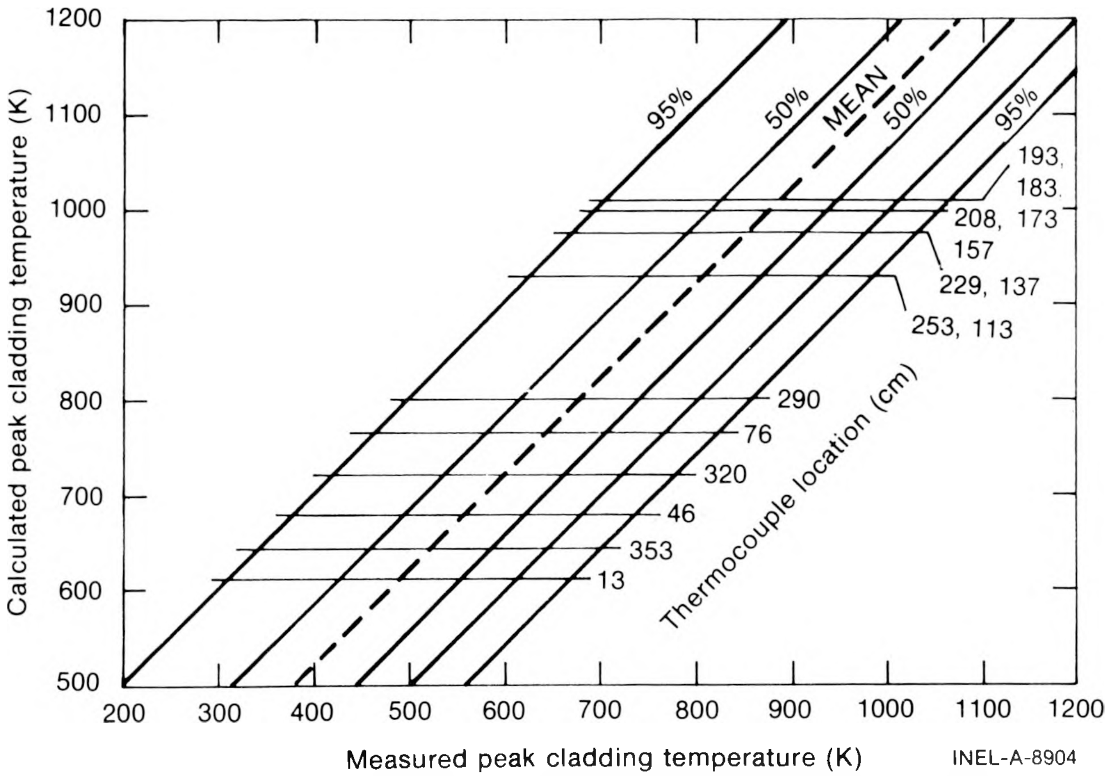

Fig. 34 Prediction intervals (at 50% and 95% confidence levels) for Semiscale Mod-3 Test S-07-1 based on all Semiscale Mod-1 Test S-04-6 data.

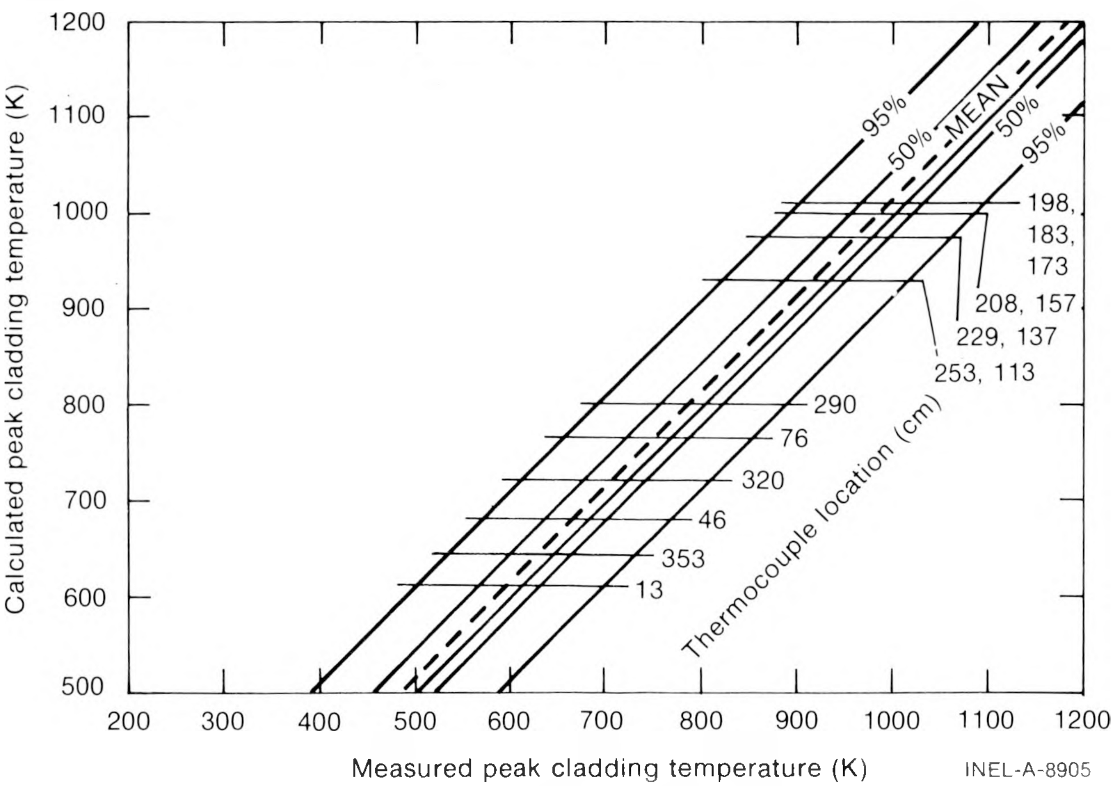

Fig. 35 Prediction intervals (at 50% and 95% confidence levels) for Semiscale Mod-3 Test S-07-1 based on early CHF data from Semiscale Mod-1 Test S-04-6.

800 runs were required to generate the predictions and corresponding sensitivity studies for the 200 rods and 400 tubes considered.

Systematic processing of test data and code input/output information has been applied for specialized analysis of different fuel behavior mechanisms. More detailed data comparisons and diagnostic analyses were set up and performed to aid in the interpretation of results for the largest sample data categories; namely, pellet-cladding interaction (PCI) failure probability, transient fuel temperature response, and tube burst conditions. Generic tape files were constructed in these cases to facilitate efficient handling of measured and calculated values and the corresponding rod design, operating parameters, and plot requirements.

3. TECHNICAL SURVEILLANCE OF NRC/INDUSTRY COOPERATIVE PROGRAMS

T. R. Charlton and W. S. Haigh

The role of the Code Verification and Applications Program as technical advisor to NRC is to ensure that the data from the industry cooperative safety experimental programs are adequate for verification of LOCA analysis codes. The industry cooperative experiments are the BWR-BD/ECC (boiling water reactor-blowdown/emergency core cooling) program and the FLECHT-SEASET (full length emergency cooling heat transfer-separate effects and systems effects tests) program.

3.1 BWR-BD/ECC Program

RELAP4/MOD6 code/data comparisons have been made for Tests 6004 and 6005 of the BWR-BD/ECC program[43]. The hardware designation used for Tests 6004 and 6005 is TLTA-3 (Two Loop Test Apparatus-3). The TLTA-3 hardware represents the first TLTA configuration to simulate a BWR/6 configuration. The hardware consists of an 8 x 8 electrically heated bundle contained within a pressure vessel which is a 1:624 volume, mass, energy, and flow rate scale of a BWR. The initial thermodynamic conditions and power levels of Tests 6004 and 6005 are equal to those of an average and peak rod bundle, respectively, in a BWR/6.

A comparison of the experimental data and the RELAP4 calculations for Tests 6004 and 6005 indicates close correlation for most of the major hydraulic events. Only a qualitative comparison can be made since the data are unverified and without error bands. The comparisons of calculated values and data for the vessel depressurization, annulus mass inventory, and jet pump flows indicate good agreement. For example, the calculation predicted the jet pump suction uncover time within 0.4 second and the recirculation line suction uncover time within 1.1 senconds of the data for both tests. However, the calculated start of lower plenum flashing varied as much as 2.6 seconds from the experimental data.

The greatest differences between the calculations and the experimental data were in the heater rod temperatures. Test 6004 data showed a peak temperature of 619 K during blowdown (0 to 27 seconds) at the 2.54-m elevation, whereas the calculated peak temperature during blowdown for Test 6004 was 680 K at the 3.05-m elevation. Similarly, the Test 6005 data showed a peak temperature of 770 K during blowdown (at the 3.05-m elevation), whereas the calculated peak temperature was 1040 K at the (2.54-m elevation).

As shown in Figures 36 and 37 the calculated peak cladding temperature is above the measured value in all cases. This is because early CHF was predicted without subsequent rewet, whereas the data show delayed CHF and rewets shortly after lower plenum flashing.

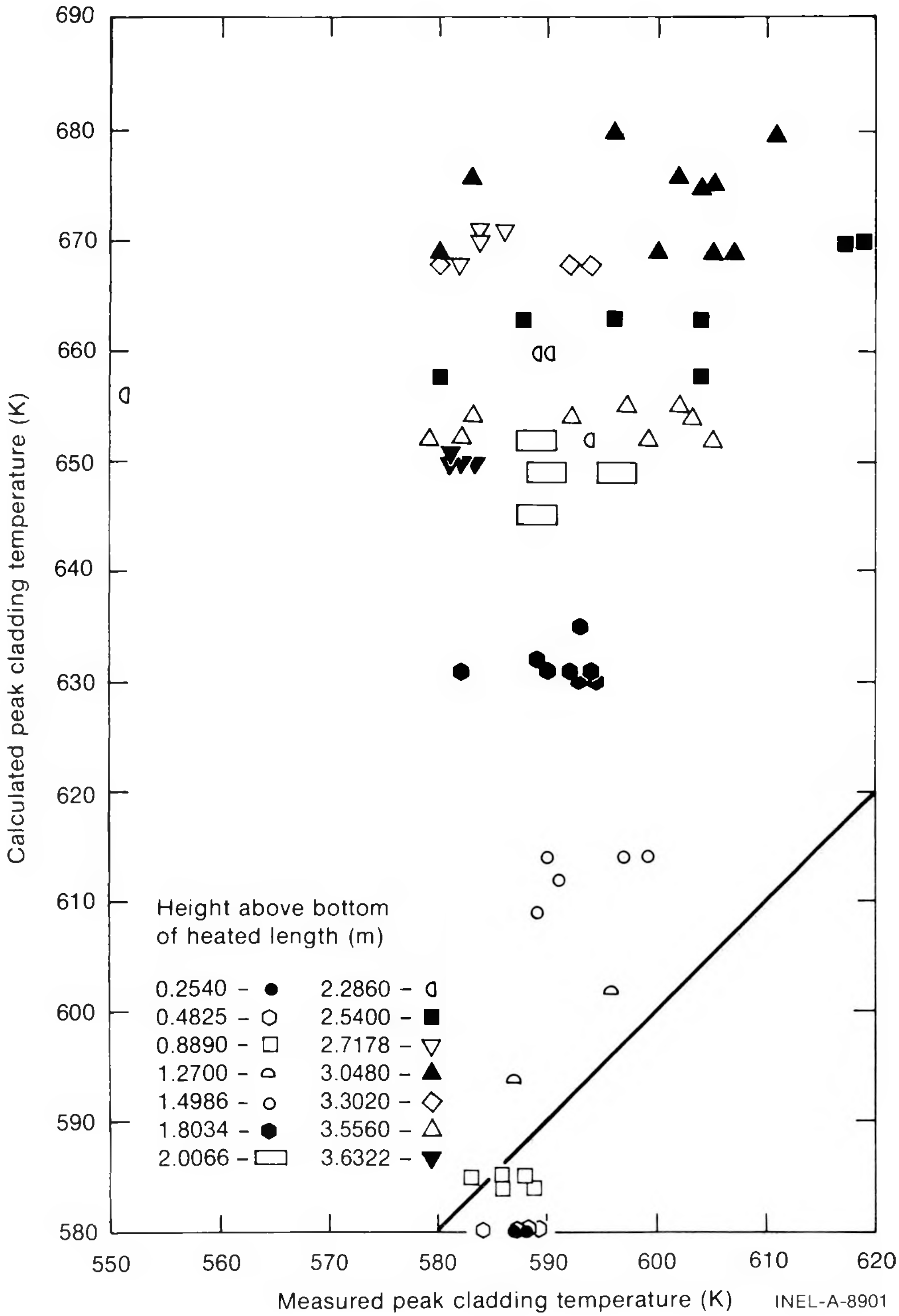

Fig. 36 Comparison of calculated and measured peak cladding temperatures for Test 6004 of the BWR-BD/ECC program.

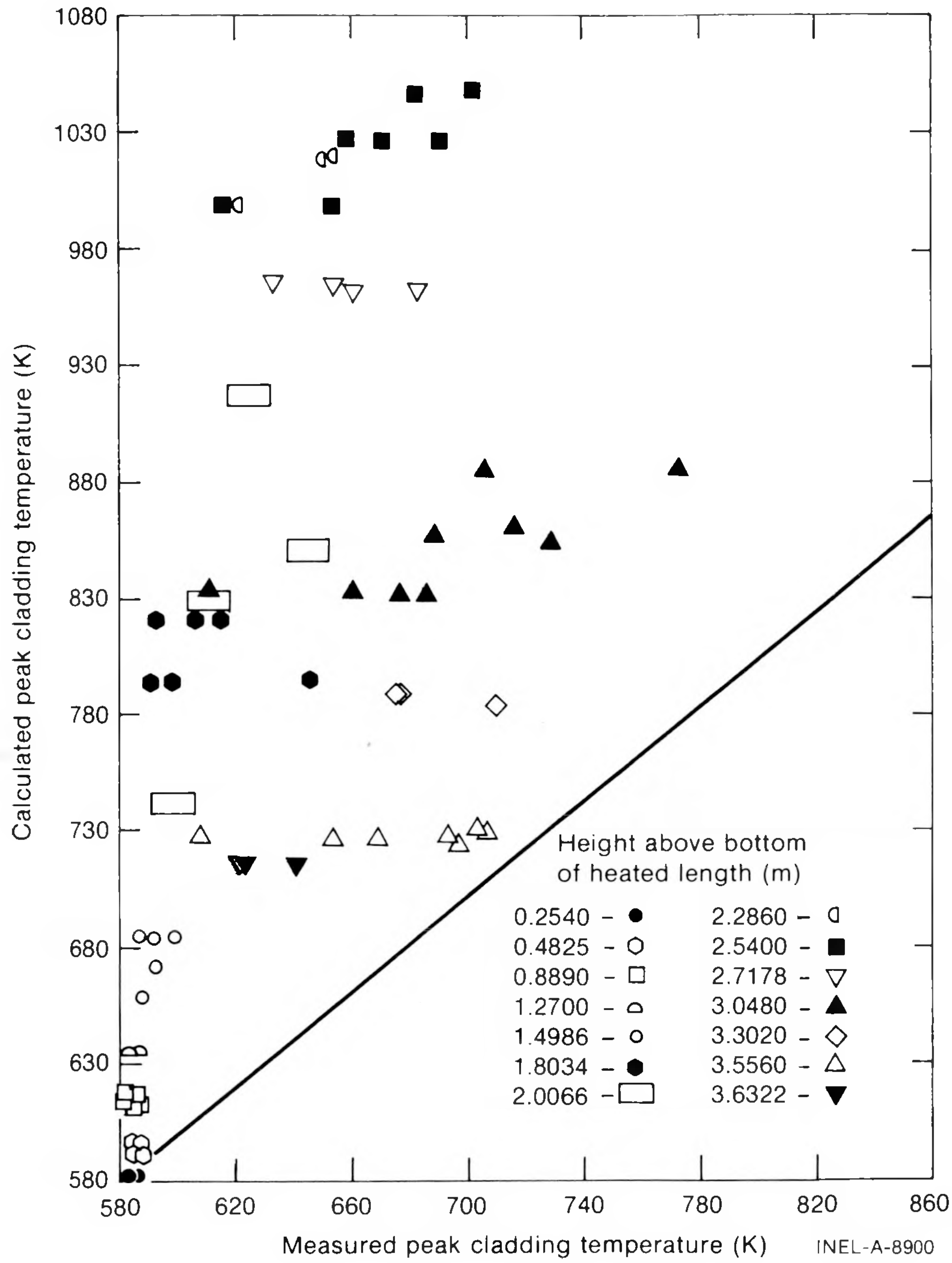

Fig. 37 Comparison of calculated and measured peak cladding temperatures for Test 6005 of the BWR-BD/ECC program.

3.2 FLECHT-SEASET Program

The major activity for the FLECHT-SEASET program[44] has been the development of a RELAP4/MOD6 model for the first steam generator test.

4. NRC/RSR DATA BANK PROGRAM

G. L. Schultz and S. F. Bankert

NRC has established the NRC/Reactor Safety Research (RSR) Data Bank Program to provide the means for collecting, processing, and making available reactor safety experimental data. These data will be collected from both foreign and domestic sources and processed on digital tape in a standard format. Copies of these digital tapes will be made available upon request. These data will also be stored on the INEL computer system.

Through June 1978, several experimental facilities have been established as sources of reactor safety experimental data. These sources include:

(1) LOFT Facility, EG&G Idaho, Inc. (INEL)

(2) Semiscale Facility, EG&G Idaho, Inc. (INEL)

(3) Moss Landing Facility, General Electric Company (Castroville, California)

(4) Westinghouse Canada Limited Facility, Atomic Energy of Canada Limited (Hamilton, Ontario)

(5) Thermal Hydraulic Test Facility (THTF), Union Carbide Corp. (ORNL)

(6) FLECHT programs, Westinghouse Electric Corp. (Pittsburgh, Pa.).

Table V shows the data from these sources currently on magnetic tape and available from INEL. In addition to the current data sources,

TABLE V

DATA IN THE NRC/RSR DATA BANK[a]

Facility	Test Identification
Semiscale	S-01-4A
Semiscale	S-01-6
Semiscale	S-02-5
Semiscale	S-02-9
Semiscale	S-03-D (forced feed reflood)
Semiscale	S-03-5 (gravity feed reflood)
Semiscale	S-04-5 (baseline ECC)
Semiscale	S-04-6 (baseline ECC)
Semiscale	S-06-1 (LOFT counterpart)
Semiscale	S-06-2 (LOFT counterpart)
Semiscale	S-06-6 (LOFT counterpart)
LOFT	L1-4 (Digital only)
AECL	CL
GE-Moss Landing	ML
Semiscale pump characteristics	Test 73, Single-phase
Semiscale pump characteristics	Test 73, Two-phase

[a] This table represents 7.4 million words of data which were compacted to 3.6 million words.

other experimental facilities have been identified as future data sources. These include:

(1) Two-Loop Test Apparatus (TLTA), General Electric Company (San Jose, California)

(2) PKL, Kraftwerk Union (Erlangen, West Germany)

(3) BWR reflood facility, AB Atomenergi (Studsvik, Sweden)

(4) Marviken Power station, AB Atomenergi (Marviken, Sweden)

The Data Bank Processing System (DBPS) is a collection of computer programs on the INEL computer system. These programs have been developed to provide the capability of accepting data from established data sources, outputting these data to tape in the standard format, adding these data to the INEL computer system, and allowing on-line selective retrieval and comparison of these data to those who have access to the INEL computer system.

The on-line users of the DBPS can be either local (INEL) or remote. For example, the NRC office at Silver Springs, Maryland, has a computer terminal which accesses the INEL computer system and the reactor safety data. With NRC approval, other remote users can be established.

Reactor safety experimental data are continually being collected from existing and new data sources and are made available to those who desire this information, either on standard data tapes or through the INEL computer system. The DBPS is being expanded to allow those with access to the INEL computer system more powerful data processing presentation and analysis tools.

VI. 3-D EXPERIMENT PROJECT

R. E. Rice, Manager

During the past quarter, the 3-D Project has been involved in instrumentation development, small-scale experimentation, and NRC technical staff support. The instrumentation projects involve specialized flow metering devices intended for use in experiments to be conducted in Japan and West Germany. The overall purpose of these experiments is the verification of the three-dimensional TRAC computer code for PWR reflood modeling. Small scale experimentation has consisted of an air-water simulation of upper core support plate and upper plenum hydraulics during the reflood phase of a LOCA. Technical staff support to the NRC has included a number of tasks in the fields of analysis and design.

1. INSTRUMENTATION DEVELOPMENT

M. M. Hintze

The instrumentation tasks in progress include the design and construction of instrumented spool pieces and liquid level detectors for German and Japanese reflood system experiments, upper plenum turbine flowmeters for the German experiment, and downcomer drag discs for the Japanese experiment. These experiments are subscale models of complete nuclear steam supply primary systems. The tests have electrically heated cores of 340 rods (German) and 2000 rods (Japanese), with the remainder of the systems scaled proportionally. They will be tested under conditions simulating the reflood phase of a loss-of-coolant accident. Preliminary design of the instrumented spool pieces has been completed, and machining of the spool pieces has begun. Facilities for two-phase steam/water operational testing and calibration of prototype spool pieces have been contracted with Wyle Laboratories of Norco, California. Design of the liquid level detector for the Japanese experiment to be performed by the Japan Atomic Energy Research Institute (JAERI) has been completed and final fabrication is in progress. Some components of the liquid level detector assemblies (support tubes and brackets) have been finished and shipped to Japan. These liquid level detectors will be installed in the core, downcomer, and lower plenums of the reactor vessel of the Japanese reflood system experiment to assess liquid inventories in those regions during the experiments. Preliminary design and review of the downcomer drag disc project has been completed. The downcomer drag discs will be installed around the periphery of the bottom of the downcomer of the reactor vessel of the reflood system experiment in order to measure the liquid flow between the downcomer and lower plenum. The liquid level detector project (for the German experiment) and the upper plenum turbine project have completed project planning, and long-lead procurement has begun. The liquid level detectors for the German experiment serve the same purpose as those for the Japanese experiment. The upper plenum turbine meters are intended to assess flow patterns in the upper plenum, which will provide data for verification of 3-dimensional codes that are being developed to describe such behavior.

2. AIR-WATER TESTS

C. M. Mohr

The 3-D Air-Water Upper Plenum Tests are being performed to provide preliminary data concerning the flow mechanisms of importance during the reflood phase of a loss-of-coolant accident. Of eight scheduled test series, four have been completed. The remainder will be performed during the next quarter. In addition to these tests, additional testing was undertaken to better understand the capabilities of the test apparatus. The additional experiments consisted of three test series to determine:

(1) The effect of a poorly distributed air flow on countercurrent flooding behavior

(2) The behavior of the air-water mixer used to simulate core water entrainment

(3) The effect of entrained water in the air flow on the countercurrent flooding behavior.

The results of the first four scheduled test series are given in Figure 38. The curves are given in terms of the gas and liquid Kutateladze numbers

$$K_\ell = j_\ell \rho_\ell^{0.5} / [g\sigma(\rho_\ell - \rho_g)]^{0.25}$$

$$K_g = j_g \rho_g^{0.5} / [g\sigma(\rho_\ell - \rho_g)]^{0.25}$$

where

j = volumetric flux through flooding flow area (m/s)
ρ = density (kg/m^3)
g = gravitational constant (9.8 m/s^2)
σ = surface tension (N/m)

and g and ℓ subscripts refer to the gas and liquid phases respectively.

From these data, the following conclusions have been drawn:

(1) The flooding behavior observed is strongly geometry dependent

(2) While the linear behavior observed in classical flooding experiments still occurs when gas and liquid flows are plotted using the dimensionless groups shown in Figure 38, the results differ markedly from those predicted by empirical correlations based on classical flooding experiments.

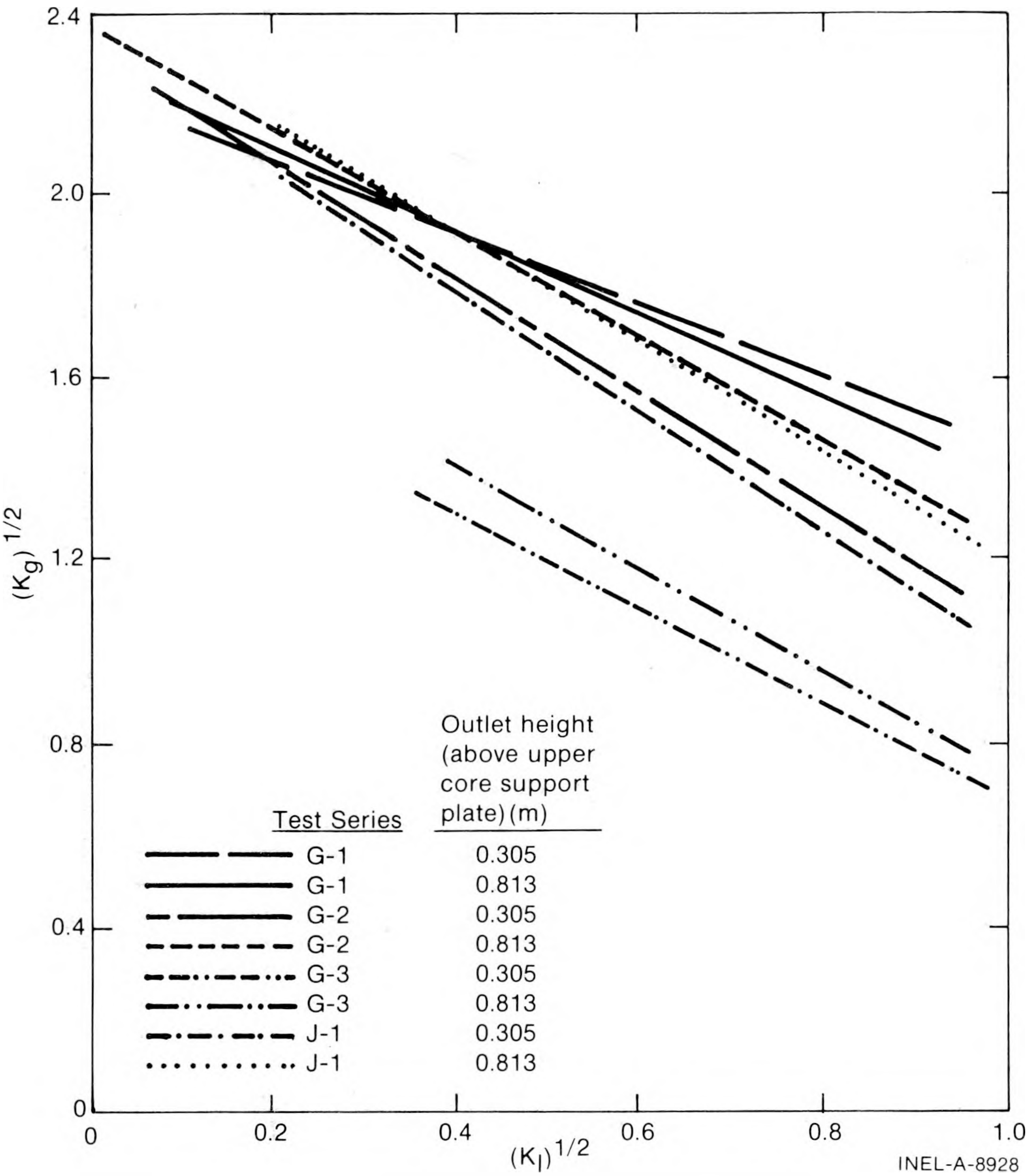

Fig. 38 Comparison of square root of Kutateladze numbers (liquid versus gas) for the first four series of 3-D Air-Water Upper Plenum Tests.

Countercurrent flooding data gathered in simple (annulus or single tube) geometries consistently display lower flooding curves than those shown in Figure 38. This is viewed as a further effect of geometry.

Experiments by Hagi, Wallis, and Richter[45] have led to the conclusion that more complicated flow paths lead to an increase in flooding, which is in agreement with the trend indicated by the data from the first four 3-D test series. Completion of this project is planned for the following quarter, including preparation of a final report documenting all phases of the experiment program.

3. TECHNICAL SUPPORT

R. A. Livingston

Technical support to the NRC has continued principally through participation in meetings with the German and Japanese 3-D Program representatives. Analytical and design tasks developed as a result of these meetings have included recommendation of upper plenum internal arrangement for the Japanese slab-core experiment and coordination of instrumentation requirements with other NRC consultants. In addition, as a follow-on to work performed previously, a detailed report of alternative slab-core design was issued, along with detailed supporting analysis.

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